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Sampling and Analysis Plan for the 233-S Plutonium Concentration Facility

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Department of Energy

CONCURRENCE PAGE

Title: Sampling and Analysis Plan for the 233-S Plutonium Concentration Facility

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Sampling and Analysis Plan for the 233-S Plutonium Concentration Facility

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United States Department of Energy

P.O. Box 550, Richland, Washington 99352

EXECUTIVE SUMMARY

This sampling and analysis plan (SAP) provides information and instructions to be used for sampling and analysis activities in the 233-S Plutonium Concentration Facility. The information and instructions herein are separated into three parts:

PART I Provides project background information, an updated list of contaminants of concern and related action levels, the plan for project task action, and two separate data quality objectives (DQO) summary reports. The first DQO report presented is for the chemical and radiological designation of wastes (without the use of nondestructive assay [NDA] services). The second DQO report presented is for the use of NDA services for the radiological designation of wastes.

PART II Provides the quality assurance project plan (QAPjP), including the activities and guidelines to provide data of known and appropriate quality.

PART III Provides field procedures and instructions to ensure representative data of known quality.

The DQO summary reports (Part I) describe the planning approach for defining the data collection design criteria for data obtained through sampling and analysis, direct-reading radiological surveys, and NDA. The DQO process is performed in accordance with BHI-EE-01, *Environmental Investigations Procedures,* Procedure 1.2, "Data Quality Objectives."

The QAPjP (Part II) presents the objectives, functional activities, methods, and quality assurance/quality control procedures associated with the 233-S Facility decontamination and decommissioning (D&D) sample collection, laboratory analyses, radiological surveys, and onsite analyses (NDA). The QAPjP follows U.S. Environmental Protection Agency guidelines contained in *EPA Requirements for Quality Assurance Project Plans for Environmental Data Operations* (EPA 1994a).

The field sampling plan (Part III) provides instructions for sample collection, laboratory analyses, radiological surveys and onsite analyses (NDA) during D&D activities at the 233-S Facility. Data collection will be used to identify the chemical, hazardous, and radiological contamination of the facility structure and internal components and the wastes resulting from the D&D activities and will support the preparation of the waste profile summaries to determine the appropriate waste disposition in accordance with *Washington Administrative Code* 173-303, "Dangerous Waste Regulations"; *Hanford Site Solid Waste Acceptance Criteria* (FH 2001); and *Environmental Restoration Disposal Facility Waste Acceptance Criteria* (BHI 1998).

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ACRONYMS

AA	alternative action
ALARA	as low as reasonably achievable
BHI	Bechtel Hanford, Inc.
CERCLA	*Comprehensive Environmental Response, Compensation, and Liability Act of 1980*
CFR	*Code of Federal Regulations*
COC	contaminant of concern
COPC	contaminant of potential concern
CWC	Central Waste Complex
CWS	Chemical Warfare Service
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
DOE-AME	U.S. Department of Energy, Assistant Manager for Environmental Management
DOP	di-octyl phthalate
dpm	disintegrations per minute
DQO	data quality objectives
DR	decision rule
DS	decision statement
EDPI	Engineering Department Project Instruction
EPA	U.S. Environmental Protection Agency
ERC	Environmental Restoration Contractor
ERDF	Environmental Restoration Disposal Facility
FSR	field sampling request
FY	fiscal year
GTCC	greater than Class C
HASQARD	*Hanford Analytical Services Quality Assurance Requirements Document*
HEPA	high-efficiency particulate air
JHA	job hazard analysis
LLW	low-level waste
MDA	minimum detectable activity
NDA	nondestructive assay
NIST	National Institute of Standards and Technology
PCB	polychlorinated biphenyl
PFP	Plutonium Finishing Plant
PPE	personal protective equipment
PR	plutonium removal
PSQ	principal study question
QA	quality assurance
QAPjP	quality assurance project plan
QC	quality control
QS&H	Quality, Safety, and Health

RadCon	Radiological Controls
RC	recycle
RCRA	*Resource Conservation and Recovery Act of 1976*
REDOX	Reduction-Oxidation
RL	U.S. Department of Energy, Richland Operations Office
RPM	Remedial Project Manager
RWP	radiological work permit
SAF	sampling authorization form
SAP	sampling and analysis plan
SWP	special work permit
TMU	total measurement uncertainty
Tri-Party Agreement	*Hanford Federal Facility Agreement and Consent Order*
TRU	transuranic
TRUSAF	Transuranic Waste Storage and Assay Facility
TSR	technical safety requirement
WAC	*Washington Administrative Code*
WS	waste stream

PART I

DATA QUALITY OBJECTIVES SUMMARY REPORT

Provides project background information, the plan for project task action, and two separate data quality objectives (DQO) summary reports. The first DQO report presented is for the chemical and radiological designation of wastes (without the use of nondestructive assay [NDA] services). The second DQO report presented is for the use of NDA services and radiological surveys for the radiological designation of wastes.

1.0 INTRODUCTION

The sampling and analysis plan (SAP) for the 233-S Plutonium Concentration Facility (233-S Facility) has been revised to incorporate the following:

- Update the list of contaminants of concern (COCs) and the related action levels based on current project knowledge.
- Provide the necessary guidance for use and implementation of direct-reading radiological surveys and nondestructive assay (NDA) measurements in obtaining the required data for waste designation.
- Technical editing on content of SAP.

This SAP does not provide for sampling and analysis or radiological surveys to determine if waste or the site can be released from radiological controls in accordance with U.S. Department of Energy (DOE) Order 5400.5, *Radiation Protection of the Public and the Environment.* Additionally, items and equipment that are surveyed and released in accordance with BHI-RC-04, *Radiological Control Work Instructions*, Instruction 4.4, "Material Release," are not provided for in this SAP.

1.1 DATA QUALITY OBJECTIVES

The data quality objectives (DQO) process for the 233-S Facility is a U.S. Environmental Protection Agency (EPA)-recommended approach to planning and coordinating data acquisition. This process is used as a decision-making tool to assess the use of historical or previously acquired data, and also establishes interfaces and promotes communication with key decision makers and other stakeholders. These decision-makers and stakeholders include representatives from the U.S. Department of Energy, Richland Operations Office (RL) and EPA.

The primary objective of the DQO process is to establish a consistent, cooperative, and streamlined approach to plan environmental data acquisition, with an emphasis on reducing cost. The DQO process employed is based on the *Guidance for the Data Quality Objectives Process* (EPA 1994b). The DQO process involves the following seven steps:

- Step 1, Problem Statement
- Step 2, Identify the Decisions
- Step 3, Identify Inputs to the Decisions
- Step 4, Define the Boundaries of the Study
- Step 5, Develop Decision Rules
- Step 6, Specify Limits on Decision Errors
- Step 7, Optimize the Design.

1.2 PROJECT OBJECTIVES

The objective of the 233-S Facility DQO is to determine sampling/analysis and radiological survey (direct-reading and NDA) requirements during waste stream characterization to provide information for worker safety and to support waste designation and disposal decisions during decontamination and decommissioning (D&D). The characterization data will be used to prepare the waste profile summary for evaluations against waste acceptance criteria to determine appropriate disposal options.

1.3 PROJECT EXCLUSIONS

The project boundary for this DQO includes the 233-S Building and subsurface structures (to a depth of 0.9 m [3 ft] below grade). Localized contamination found below the 0.9-m (3-ft) level may be removed; however, extensive soil remediation (i.e., locating and removing extensive contamination migration) is not part of this project. Contamination remaining will be identified/documented in accordance with BHI-RC-04, Instruction 4.2, "Radiological Surveys."

1.4 PROJECT ASSUMPTIONS

- The decommissioning of the 233-S Facility is controlled under the *Comprehensive Environmental Response, Compensation, and Liability Act of 1980* (CERCLA) process.
- All radiologically contaminated material shall be disposed at the Environmental Restoration Disposal Facility (ERDF) as long as waste acceptance criteria are not exceeded.
- Transuranic (TRU)/greater than Class C (GTCC) waste and waste that exceeds the ERDF acceptance criteria will be transported to the Central Waste Complex (CWC) for storage and will meet the requirements of *Hanford Site Solid Waste Acceptance Criteria* (FH 2001), or an exception will be obtained.
- Plant components (i.e. equipment, vessels, and piping) originating from D&D of the 233-S Facility process hood will be assumed to be TRU/GTCC waste, unless determined otherwise.
- "Soft" waste (e.g. gloves, wipes, smears, tape, sleeves, and any discarded personal protective equipment [PPE]) that is generated in support of the 233-S Facility process hood D&D activities will be assumed to be low-level waste (LLW), unless determined otherwise.
- "Step-off pad" waste is made up of discarded/used PPE that is generated between the high contamination area and the contamination area, and at the boundary between the contamination area and the radiological buffer area. It will be assumed to be LLW, unless determined otherwise.

- The radiological survey data/information that is obtained as part of the radiological control (RadCon) program is acceptable to be used for verifying process knowledge and making waste management decisions.

- Sampling and laboratory analysis data obtained under the guidance of DOE/RL-97-87, Rev. 0 (DOE-RL 1998) has been used to develop the isotopic ratios (see Table 4-5) for waste and to prepare the 233-S Facility waste profile.

- Radiological control surveys performed in accordance with BHI-RC-04, *Radiological Control Work Instructions* and BHI-RC-05, *Radiological Instrumentation Instructions* implement the Environmental Restoration Contractor's (ERC's) radiation protection program (described in BHI-RC-01, *Radiation Protection Program Manual)* are as follows:

  - Document current and verify the historical radiological conditions of the facility.

  - Detect changing conditions and gradual buildup of radioactive material during D&D activities.

  - Verify the effectiveness of engineering and process controls for containing radioactive material and reducing radiation exposure.

  - Identify and control potential sources of individual exposure to radiation and/or radioactive material.

  - Make waste designation determination/decisions.

- NDA measurements, when used for waste designation, will be obtained for each individual waste item to qualify the item for disposal at ERDF. NDA measurements for waste items that do not qualify for disposal at ERDF may be obtained by individual measurement or by measurement of barrels, boxes, or drums of such waste items.

1.5 EXISTING REFERENCES

Table 1-1 presents a list of all of the references that were reviewed as part of the scoping process and a summary of the pertinent information contained within each reference. These references are the primary source for the background information presented in Section 1.4.

Table 1-1. Existing References Used for DQO Scoping Process.

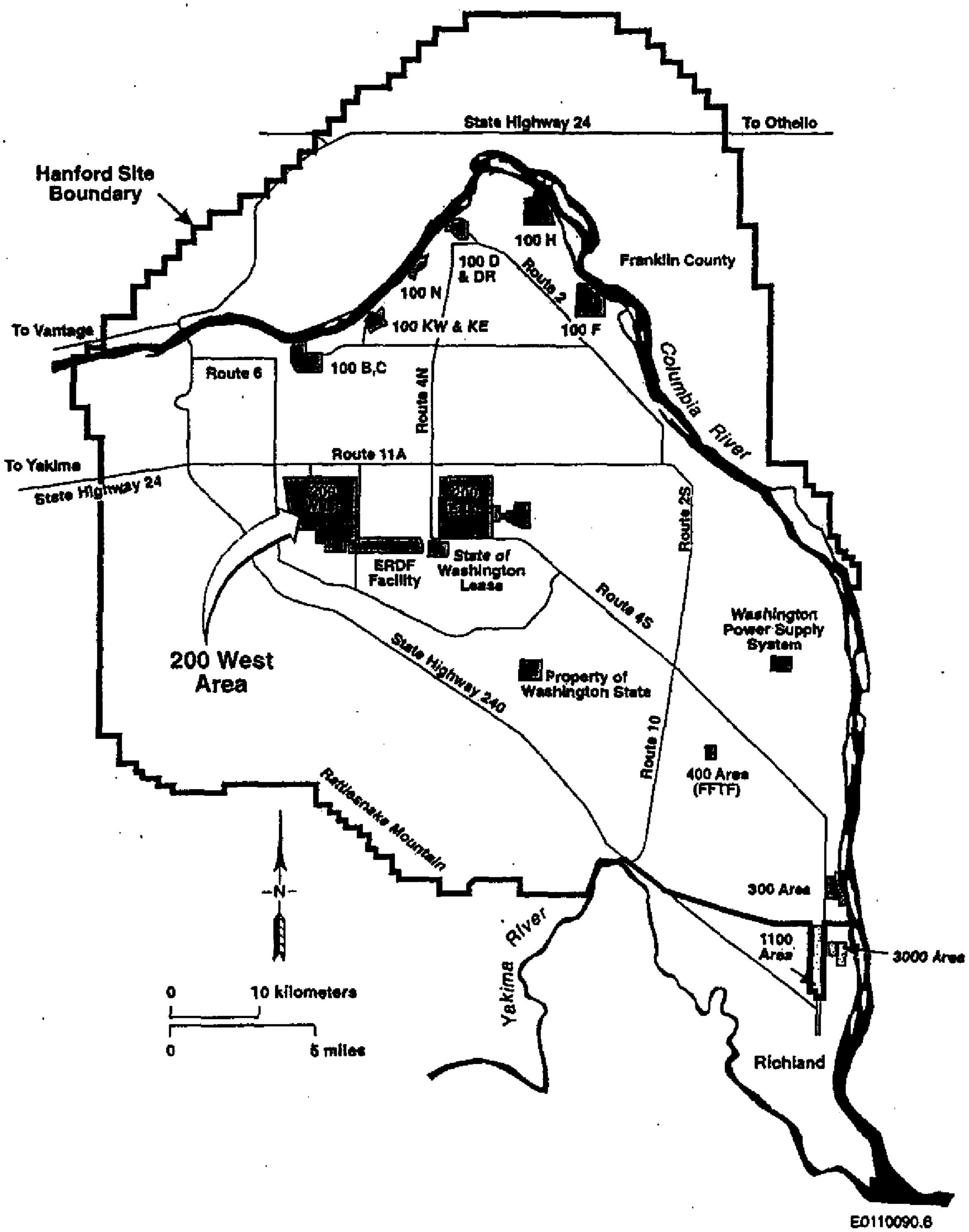
Reference	Summary
Sampling and Analysis Plan for the 233-S Plutonium Concentration Facility, DOE/RL-97-87, Rev. 0 (DOE-RL 1998)	Provides guidance for sampling and analysis of materials from the 233-S Facility prior to issuance of Rev. 1 of this SAP. Contains comprehensive information on facility description, historical data, process history and extent of contamination within the facility, list of waste steams, waste matrices, and contaminants of concern.
Criticality Safety Program Requirements for 233-S, 0233-DB-G0005, Rev. 3 (BHI 2001d)	Provides the criticality safety TSRs and controls for defense-in-depth for fissionable material and waste containers. Provides NDA criteria (reported in dose rate [beta/gamma]) for process components associated with the process hood.
233-S Plutonium Concentration Facility Authorization Basis Manual, 0233S-AB-G0002, Rev. 6 (BHI 2001c)	Identifies the broad waste categories that will be generated from the D&D of the 233-S Facility.
Radiological Characterization of the 233-S Facility, WHC-SD-CP-TI-163, Rev. 0 (WHC 1990b).	Provides radiological characterization data for the process drains, exhaust ducts, concrete floors and walls, and roofing material.
233-S Facility Potential Chemical Hazards, WHC-SD-DD-TI-056, Rev. 0 (WHC 1990a)	Provides chemical characterization data for the process drains, exhaust ducts, concrete floors, and walls.
Environmental Restoration Disposal Facility Waste Acceptance Criteria, BHI-00139, Rev. 3 (BHI 1998)	Provides land disposal restriction limits and the chemical and radiological concentration limits in the wastes to be disposed at ERDF.
Hanford Site Solid Waste Acceptance Criteria, HNF-EP-0063, Rev. 6 (FH 2001)	Provides waste acceptance criteria for storage at the CWC.
Final Characterization Report for the Non-Process Areas of the 233-S Plutonium Concentration Facility, BHI-01032, Rev. 0 (BHI 1997)	Provides characterization data for the non-process piping, concrete floors and walls, paint, and roofing material.

1.6 FACILITY AND PROJECT BACKGROUND

1.6.1 Physical Description

The Hanford Site (Figure 1-1) is located in south-central Washington State and was selected as the nation's first large-scale nuclear materials production site in January 1943. Plutonium was produced by irradiating uranium fuel elements using reactors located in the 100 Areas of the Hanford Site.

Figure 1-1. Hanford Site Map.



After the fuel was irradiated, it was taken to separation plants located in the 200 Areas, where the cladding was removed from the fuel elements and plutonium was extracted. The Reduction-Oxidation (REDOX) Plant began operation in January 1952. The REDOX Plant was the world's first nuclear solvent extraction plant using the reduction-oxidation process and operated through July 1967. The 233-S Facility was built in 1955 to expand production and further concentrate the plutonium-nitrate product solution from the REDOX Facility. The 233-S Facility is located on the north side of the REDOX Plant in the Hanford's 200 West Area (Figures 1-1 and 1-2).

The 233-S Facility is composed of the original 233-S Process Building, additions/modifications, and interconnecting piping, trenches, and ducting. The 233-S Building was modified by expansion in 1958. This expansion included the addition of maintenance platforms in the process cell viewing room with an exterior stairwell and air locks for entry, an additional plutonium removal (PR) can room, and a spare exhauster. Modifications in 1962 included the installation of an anion-exchange purification process in the process hood, the conversion of one plutonium concentrator for neptunium concentration, other vessel modifications, and numerous piping modifications. The 233-SA Exhaust Filter Building was added in 1964 after a process upset in 1963 that resulted in a fire.

1.6.2 Facility Description

The 233-S Building is a reinforced-concrete structure, 11.3 m (37 ft) by 25.7 m (86 ft) with 20.3-cm (8-in.)-thick walls and 15.2-cm (6-in.)-thick floors. The building includes the main contaminated areas, primarily where process-related activities formerly took place, and non-process areas where contamination is significantly less. The main contaminated areas consisted of the process cell and viewing room and the PR can loadout room. The non-process areas consisted of two can storage rooms, a pipe gallery, the control room, the equipment room, special work permit (SWP) change room, the toilet, an abandoned filter box, and three air locks (Figure 1-3).

1.6.3 Specific Areas Within the Facility to Be Investigated

The following are the specific areas of concern within the original facility, most of which have already been investigated.

233-S Process Area

The process area is a four-story-high bay with 30.5-cm (12-in.)-thick concrete walls, and it was divided into two zones. The two zones (the process hood and the viewing room) were separated with a partition of transparent panels and structural steel, which have been removed and replaced by glove-bags. The transparent panels were previously covered with opaque paint for contamination-control purposes.

Figure 1-2. Location of the 233-S Facility.

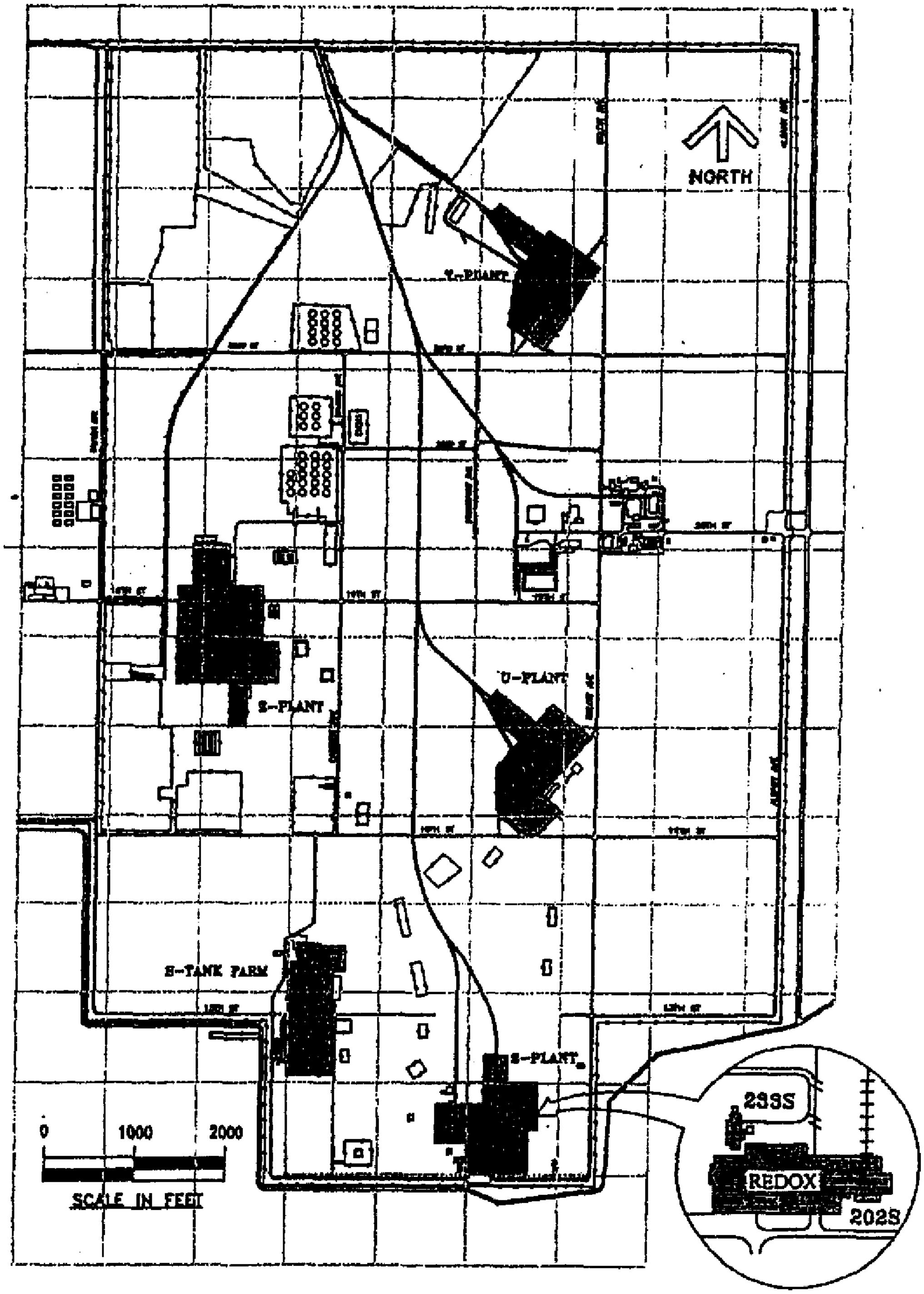


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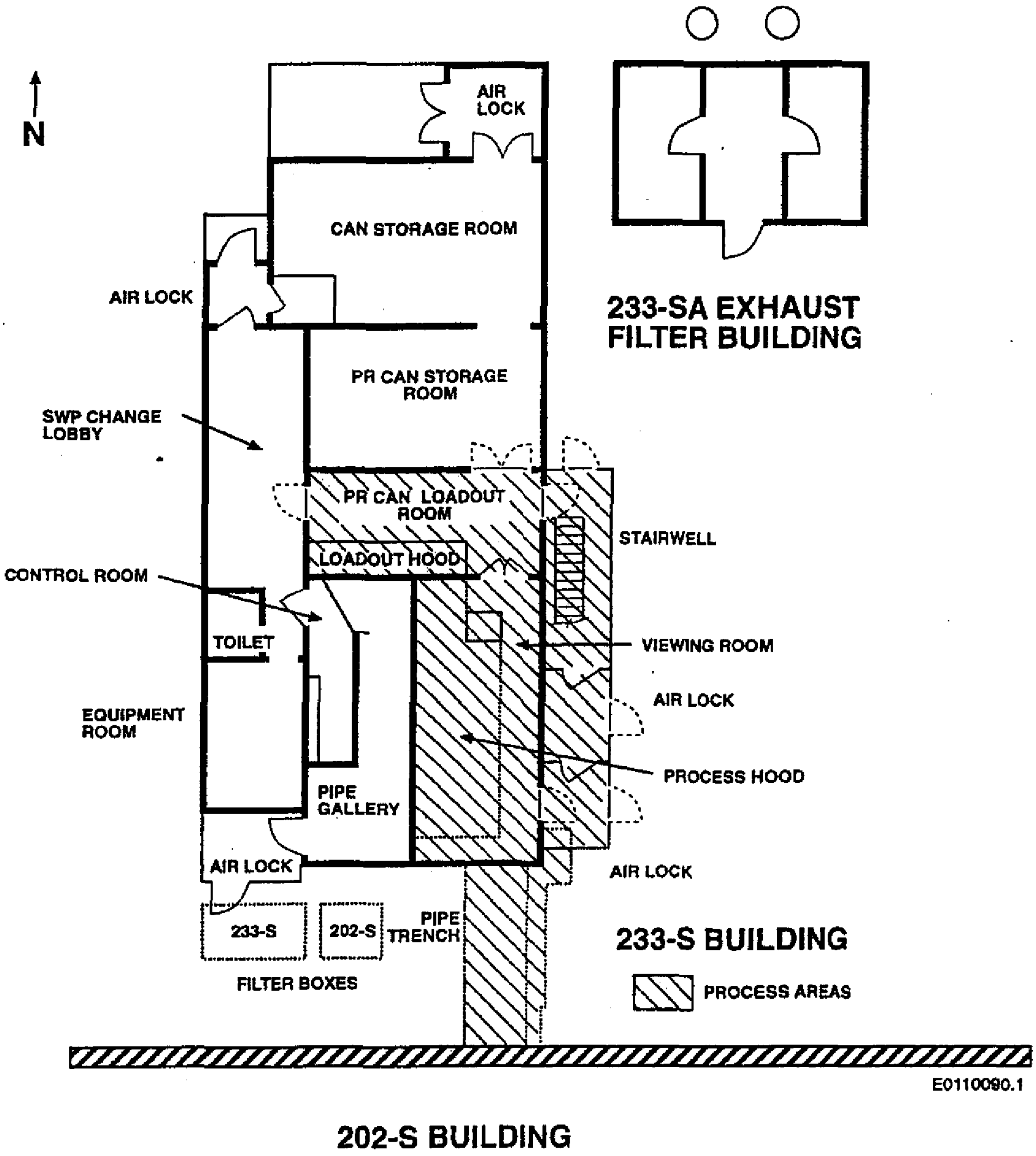
Figure 1-3. Facility Diagram.



Process Hood

The process hood is 9.7 m (32 ft) high and contained a process system array with criticality-favorable process vessels up to 7 m (23 ft) tall and 17.8 cm (7 in.) inside diameter. Plutonium nitrate solution was pumped from the REDOX E-3 feed tank to the 233-S L-12 feed tank. The solution was concentrated by boiling and/or ion exchange treatment and loaded into PR cans in the loadout hood prior to shipment for final work at the 231-Z Plutonium Isolation Building of the 234-5Z Plutonium Finishing Plant (PFP) (also referred to as Z Plant).

Viewing Room

The viewing room provides access to each of the three upper levels of the process hood via three open-grating walkways along the east and south sides of the process hood enclosure. The original access ladder remains in the southwest corner. The walkways are located to divide the height of the cell into approximately equal segments of 2.4 m (8 ft). At the north end of the process hood, the wall at the upper level supported electric and process instrumentation equipment.

PR Can Loadout

The PR can loadout and decontamination room is located on the north side of the process hood. The loadout hood was located on the south side (or common wall with the process hood) and was a confinement-type work station used for loading PR cans with concentrated plutonium nitrate solution, neptunium solution, and unloading recycle (RC) cans for rework in 233-S or 202-S. Decontamination of the PR and RC cans was performed in the loadout hood. There are no PR or RC cans remaining at the 233-S Facility.

PR Can Storage Room

The can storage rooms allowed 68 PR and RC cans to be stored while awaiting shipment or recycling back into the system. These rooms are on the north side of the loadout room.

Non-Process Pipe Gallery and Control Room

The pipe gallery contained non-process support lines from the REDOX Building that entered the area through the viewing room. Equipment in the room included instrument lines, steam lines, a chemical makeup tank, and a variety of control panels. The control panels were separated from the non-process pipe gallery by plastic panels that created an isolated control room.

Equipment Room

The equipment room contains the necessary equipment, ducting, and wiring to provide and control makeup air to the building. Much of the ducting was insulated with asbestos materials.

Abandoned Filter Box

The abandoned 233-S filter box is a reinforced-concrete structure located below grade between the REDOX Building and the 233-S Building. The filter box is approximately 1.8 m (6 ft) wide by 1.8 m (6 ft) deep by 3.65 m (12 ft) long, with 0.15-m (6-in.)-thick walls, and was used as a backup system during the time of the 1963 fire. The primary system was an above-grade filter housing that provided filtration system for the facility. A temporary filtration unit was recently installed to allow tie-in of the 233-SA Building, the unneeded ductwork and above-ground filters were removed, and the filter box was abandoned. It is unknown if the Chemical Warfare Service (CWS) filters were abandoned in place.

The abandoned 202-S filter box is similar, except that it is 1.8 m (6 ft) in all three dimensions.

Process Pipe Trench

The pipe trench is a 7.15-m (23.5-ft)-long concrete sub-grade structure, extending between the REDOX Building and the southeast corner of the 233-S Building. The original pipe trench was divided into two parallel sections to separate the radiological solution transfer lines and nonradiological piping. A neptunium pipe trench was added in the 1962 upgrade and is located adjacent to the original pipe trench.

233-SA Exhaust Filter Building

The 233-SA Exhaust Filter Building was constructed following the 1963 fire to handle the exhaust ventilation for the 233-S Facility. The 233-SA Exhaust Filter Building is a one-story, 4.9-m (16-ft) by 7.3-m (24-ft) reinforced-concrete structure with 15.2-cm (6-in.)-thick walls. The filter building is located on a 7.3-m (24-ft) square, 0.2-m (8-in.)-thick reinforced-concrete pad at the northeast corner of the 233-S Process Building. The filter building contains two parallel filter banks. Each filter bank has a series of double high-efficiency particulate air (HEPA) filters, each with its own exhaust fan, a 7.6-m (25-ft)-high metal stack, and sampling equipment. The fans and stacks are located north of the building and are designated as 296-S-7 east and 296-S-7 west.

Facility Roof Structures

The roof of the 233-S Process Building and 233-SA Filter Building consist of 0.15-m (6-in.)-thick concrete covering the building sections. The newer sections of the 233-S Process Building that are constructed with metal walls affixed to structural steel frames are roofed with metal plates. The roofs include the base structural materials (metal or concrete) and an insulation layer covered with tarred gravel. The facilities' roofs have currently been declared sufficiently sound to support minor on-roof repair operations to seal cracks and prevent water leakage into the facilities.

1.6.4 Process Description

Plutonium was produced by irradiating uranium fuel elements using reactors located in the 100 Areas of the Hanford Site. After the fuel was irradiated, it was taken to the 202-S REDOX Plant (located in the 200 West Area), where the aluminum cladding was stripped from the fuel elements and plutonium was extracted. The plutonium solution was transferred from the 202-S REDOX Building to the 233-S Facility, where the plutonium solution was concentrated and loaded into PR cans for transport to the PFP for further processing. In 1962, operations at the 233-S Facility were expanded to include a neptunium concentration and loadout process, as well as an ion-exchange plutonium purification process.

The neptunium process was similar to the initial plutonium process. Neptunium solution was received from the 202-S Building and concentrated on a batch-by-batch basis. The concentrated neptunium solution was then loaded into transfer cans and transported to another facility for further processing.

During the ion-exchange process, solutions containing plutonium and undesirable impurities were passed through a resin bed where the plutonium absorbed onto the resin, and the impurities remained in the solution and left the system. The purified plutonium was then chemically removed from the resin and loaded into PR cans for transport to the 231-Z Building at PFP.

1.6.5 Summary of Major Recorded Spills and Releases

In 1956, an air-activated diaphragm valve (located between a plutonium nitrate concentrator and a receiver vessel) failed, which allowed the acidic solution to work back through a copper air-supply line. The acidic solution corroded the copper, and about 32.5 g of plutonium solution was found in two visible spills, which showed contamination levels greater than 7×10^6 disintegrations per minute (dpm) alpha (which is an off-scale measurement on the survey meter). The ventilation system was set up to pressurize the change room and control room with respect to the process area and outside areas. Shortly after the incident occurred, it was discovered that contaminated air was also being forced outside through the gravity dampers and building doors.

In November 1963, chemical reactions occurred within the scrub-load section of the L-18 ion-exchange unit. The reaction resulted in a rapid pressure buildup within the column and a release at a flange joint, causing a pyrolytic, ejection of plutonium-loaded resin beads. The primary barrier (i.e., the piping) was breached and a plutonium/resin pyrolytic reaction ignited a fire, causing extensive damage to the process equipment. Gross alpha contamination was spread within the process area, and radiological contamination was distributed to other portions of the facility, including the exterior roof.

1.6.6 Past Decontamination and Stabilization Activities

Decontamination and decommissioning of the 233-S Facility was selected in 1978 as a demonstration project. A major effort began to decontaminate the facility, which ceased in 1981 because of insufficient funding. This activity accomplished initial characterization and housekeeping of the facility and removed the contents and equipment from the loadout hood. The contamination within the loadout hood was stabilized, and plexiglass panels equipped with HEPA filters were installed to cover the openings. Subsequently, the interior of the loadout hood was recontaminated by particulate migration through the previously sealed wall penetrations from the adjoining process hood. The hood contains a sump (45.7 cm by 45.7 cm by 10 cm) [18 in. by 18 in. by 4 in.).

Stabilization activities, including interior and exterior areas of the facility, were completed in 1987. The stabilization work sealed the joints around cover blocks over the 202-S REDOX column laydown trench and the pipe trench between 233-S and 202-S. The activities also fixed contamination around these trenches and the north wall of REDOX with an asphalt emulsion and accomplished decontamination and fixative application inside the 233-S Building.

Since 1996, D&D activities have been ongoing at the 233-S Facility, with all non-process areas being decommissioned and stabilized for eventual demolition of the facility. The L-1A acid supply tank (located in the non-process pipe gallery) has been isolated, contamination has been fixed, and the tank is ready for removal after the roof is removed (or a large access hole is made for the removal). The cold, hot, and neptunium pipe trenches have had all concrete cover blocks, piping and debris removed, remaining contamination fixed, and the trenches grouted. All original ventilation ducting has been removed, with temporary ducting being used to support D&D activities within the facility. The loadout room has been decommissioned and residual contamination has been fixed to prepare for demolition. The loadout hood and its sump have been removed.

All piping, instrumentation, and other items (except for the grated walkways) have been removed from the viewing room. All panels and accessible horizontal-channel iron supports for the panels surrounding the process hood have been removed. Loose debris on the floor of the process hood has been removed, remaining gross contamination has been removed, and a fixative has been applied to all surfaces within the process hood. Nine of the fifteen vessels and their associated piping within the process hood have been removed. Work scheduled during the remainder of the ERC's contract includes removal of the remaining vessels and piping within the process hood, the viewing room's grated walkways, and the structural steel frame of the process hood.

Follow-up work for the next contractor will include decommissioning of the 233-SA Filter Building and two below-grade filter boxes (located between 202-S and 233-S), demolishing the structures (i.e., 233-S, 233-SA, pipe trenches, and filter boxes) to 0.9 m (3 ft) below grade, and performing final status surveys.

Figure 1-4 shows the remaining vessels in the 233-S Facility process hood, and Figure 1-5 depicts the floor area and current D&D status of the facility.

Figure 1-4. Remaining Vessels in 233-S Facility Process Hood (at the End of Fiscal Year 2001).

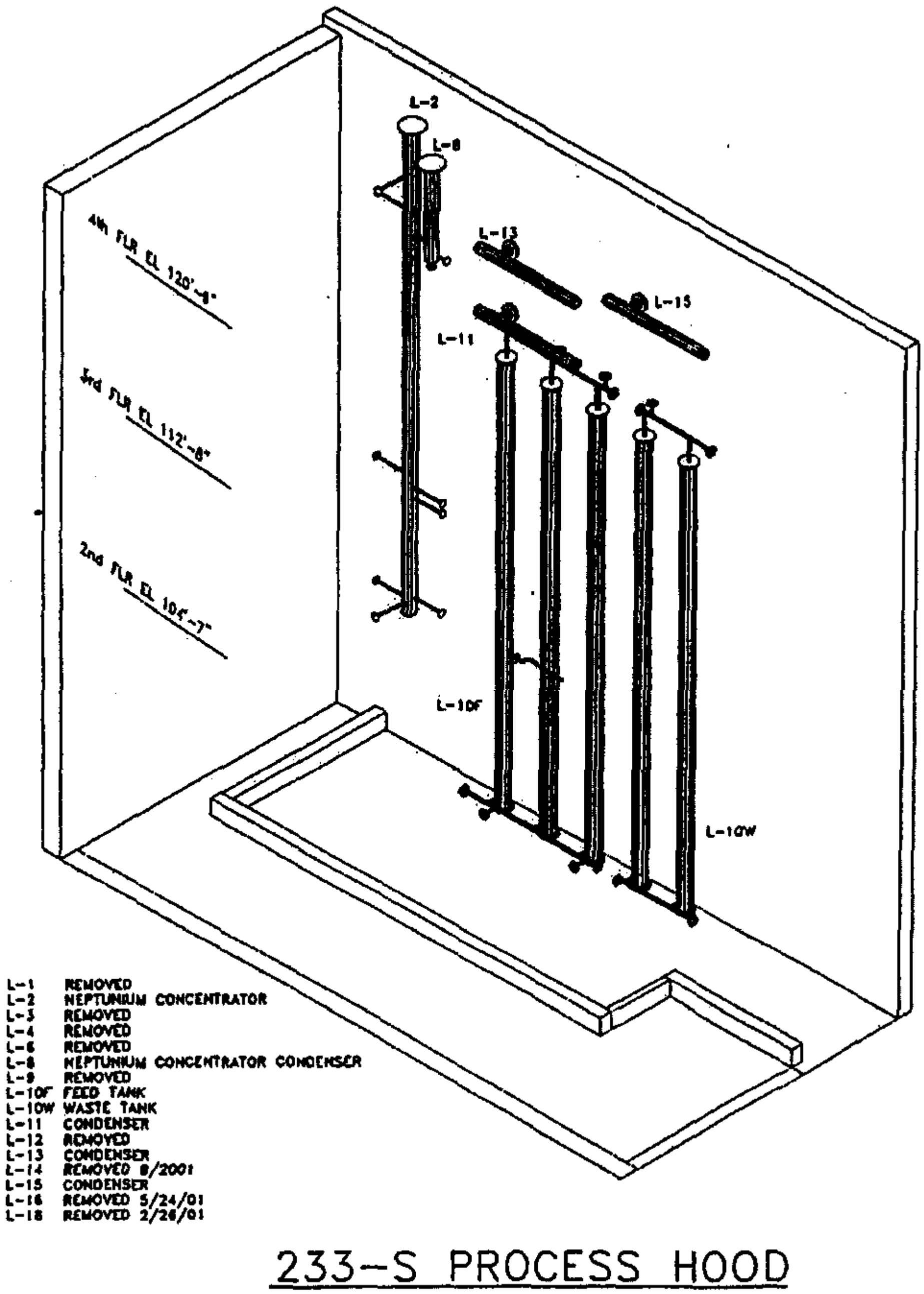
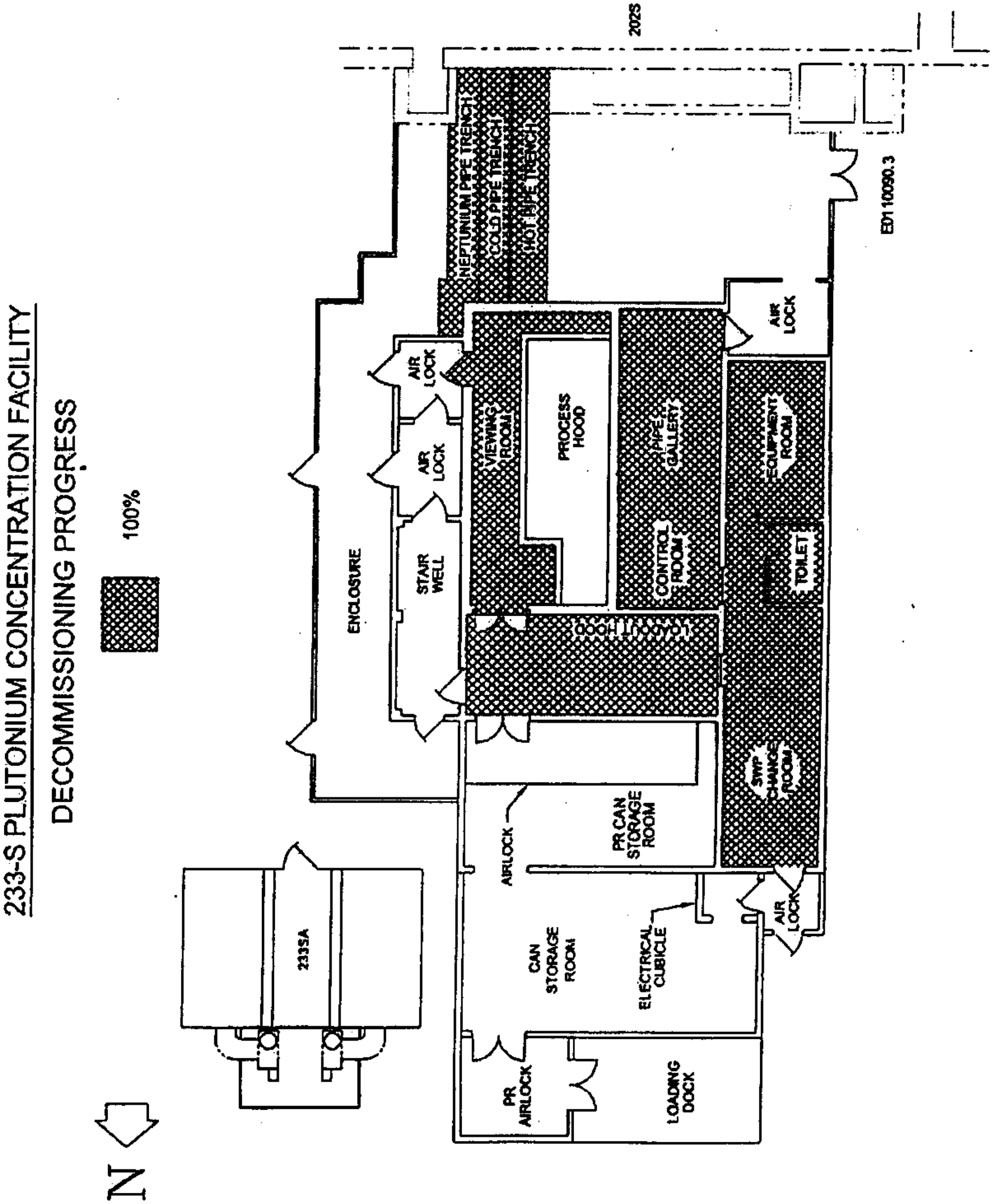


Figure 1-5. Floor Area of 233-S Facility Showing D&D (at the End of Fiscal Year 2001).



1.7 SUMMARY OF EXISTING DATA

1.7.1 Characterization

In 1990, radiological and chemical characterization surveys were performed and reported in *Radiological Characterization of the 233-S Facility* (WHC 1990b) and in *233-S Facility Potential Chemical Hazards* (WHC 1990a). The extent of alpha contamination throughout the process area currently ranges from 1×10^3 dpm/100 cm^2 removable to 6×10^8 dpm/100 cm^2 fixed and removable (process hood).

In 1996, characterization activities were conducted to evaluate the radiological status of the 233-S Facility and identify hazardous substance locations (BHI 1996). The characterization activities were limited to the non-process areas, which include the SWP change room, toilet, equipment room, electrical cubicle, control room, and pipe gallery. Radiological surveys and sample collection activities were conducted to identify the removable, residual, and radiological concentrations throughout the non-process areas. The extent of alpha contamination throughout the non-process area currently ranges from nondetectable to 1×10^4 dpm/100 cm^2 removable.

Summaries of characterization data obtained in accordance with Rev. 0 of the 233-S Facility SAP (DOE-RL 1998) are contained in the *233-S Plutonium Concentration Facility FY 1998 and FY 1999 Interim Status Report* (BHI 1999) and the *233-S Decommissioning Project Fiscal Year 2000 Status Report* (BHI 2000a).

1.7.2 Criticality Evaluation

Appendix G of the *Removal Action Report for the 233-S Plutonium Concentration Facility* (BHI 2000e) was originated by a criticality safety engineer from Bechtel National, Inc. Appendix G was further modified by the Bechtel Hanford, Inc. (BHI) criticality safety engineer before submission as part of the removal action report and authorization basis. The authorization basis package was approved by RL on September 30, 1997 (Holten 1997), for BHI to start with the non-process area piping and equipment removal activities. This was subsequently expanded to include the start of dry cleanup operations in the process hood.

During September 1999, higher concentrations of plutonium than expected were discovered in the process hood, which delayed work while the authorization basis was modified. Calculation 0200W-CA-N0016, *Criticality Analysis of 233-S Process Hood Floor and Sump* (BHI 2000b), was part of this work. Work resumption was approved by RL on January 19, 2000 (Klein 2000). During this process, eight technical safety requirements (TSRs) were developed by BHI to ensure criticality safety compliance with the double contingency criterion, as required by DOE orders and national consensus standards. The TSRs were also approved by RL (Appendix 1 of Klein 2000).

BHI calculations and documents have shown that all fissionable material in the 233-S Facility (particularly the plutonium in the process hood) will remain subcritical under all normal and credible abnormal conditions. This has been demonstrated for the process hood's individual vessels and pipes, the entire array, and the floor and sump. For all cases analyzed, sufficient criticality safety margins are maintained for which BHI concludes that criticality is not credible. In addition to the eight current TSRs, a substantial number of defense-in-depth controls are included in the design basis requirements document (BHI 2001d) to ensure that criticality remains not credible and unplanned redistribution of plutonium is prevented.

1.8 CONTAMINANTS OF CONCERN

A list of the COCs for the 233-S Facility under investigation was generated by initially listing all of the contaminants of potential concern (COPCs) based on historical process operations. Certain COPCs were then removed from the list if they were being addressed under either a separate SAP or waste management plan. Certain COPCs may also have been removed if they have a short half-life, are not regulated, pose no risk, are non-toxic, or if process knowledge/analytical data confirm that insignificant releases have occurred.

1.8.1 Total List of Contaminants of Potential Concern

Table 1-2 identifies all of the COPCs for each of the waste streams expected to be generated within each functional area of the facility for disposal. The waste streams (WSs) are numbered for tracking purposes. These waste stream numbers do not represent waste code numbers.

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
1	Process drains	Chemical flow process, spills	Deposits in pipe system/possible liquids	RCRA metals, PCBs, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOAs/SVOAs and nitric acid (pH) also become COCs.	The process drain that ran through the pipe trench and came from the process and loadout hood was characterized and removed. The drains in the floor of the loadout room and the viewing room are grouted shut and are planned to be sampled as the building is demolished.
2	Exhaust ducts	Process solution blow-back incident, chemical fire	Deposits and particulate entrainment in duct work	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^-	Original exhaust ducts have been removed and disposed. This waste stream no longer exists. Temporary exhaust ducts used during D&D remain.
3	Elemental lead	Shielding material (shielding use, chemical fire)	Lead sheeting	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^-	There is still some elemental lead remaining in the facility that is being used for shielding.
4	Anion ion exchanger (L-18)	Purification process	Spent resin/possible liquids	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	The vessel has been sampled and removed from the 233-S Facility.
5	Neptunium concentrator (L-2)	Process use	Scale, residue/possible liquids	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	This vessel remains within the process hood and needs to be sampled in accordance with the original SAP (DOE-RL 1998).

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
6	Product concentrator (L-3)	Process use	Scale, residue/possible liquid	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	The vessel has been sampled and removed from the 233-S Facility.
7	Neptunium concentrator condenser (L-8)	Process use	Scale, residue/possible liquid	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	This vessel remains within the process hood and needs to be sampled in accordance with the original SAP (DOE-RL 1998).
8	Process hood (sump)	Process use	Scale, residue/possible liquid	RCRA metals, PCBs, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOAs/SVOAs and nitric acid (pH) also become COCs.	The sump contents were sampled in accordance with the original SAP (DOE-RL 1998).
9	Process pipe trench (concrete structure)	Chemical process flow	Scabble/concrete pieces	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^-	The concrete needs to be sampled prior to disposal.
10	Process pipe trench (process pipe)	Chemical process flow	Deposits in pipe system/possible liquids	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^-	The pipe contents were sampled in accordance with the original SAP (DOE-RL 1998) and the pipes have been removed.

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
11	Non-process piping	Process solution blow-back incident, chemical fire	Deposits in pipe system/possible liquids	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, PCBs, and NO_3^-	The pipe contents were sampled in accordance with the original SAP (DOE-RL 1998) and the pipes have been removed.
12	Loadout hood (sump)	Process use, chemical fire	Scale, residue/possible liquid	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, PCBs, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	The sump contents were sampled in accordance with the original SAP (DOE-RL 1998) and the sump has been removed.
13	BF_3 tubes	Chemical fire	Contained gas	F, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs	One BF_3 tube was found and removed from the process hood. Arrangements are in process for stabilization of the tube.
14	HEPA filters	Ventilation filtration	Filter media, exhauster	RCRA metals, di-octyl phthalate (DOP), NO_3^-, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, and Co-60	HEPA filters remain in the 233-SA Building and will need to be sampled. The two below-grade filters/boxes need to be characterized.
15	Asphalt	Process solution blow-back, chemical fire	Asphalt	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, PCBs, and NO_3^-	The asphalt surrounding the facility needs to be characterized.
16	Concrete (floor/walls)	Chemical fire, process solution blow-back	Cement matrix	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, PCBs, and NO_3^-	The concrete needs to be characterized prior to demolition.

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
17	Asbestos-containing material	Asbestos fibers and chemical fire, process solution blow-back incident	Pipe insulation, cement wall board, floor tiles, valve gaskets, roofing material, duct work	Asbestos fibers, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs	All suspect ACM has been sampled in accordance with the original SAP (DOE-RL 1998) and removed. Additional ACM may be discovered during demolition of the facility.
18	Wood/ sheetrock	Chemical fire, process solution blow-back incident; RCRA metals in paint pigment	Wood matrix/sheetrock matrix	RCRA metals, Cr (total), Ni (total), Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs	None currently identified in the original structure; however, temporary structures to support D&D may require sampling.
19	Roofing material	Chemical fire	Tar, rolled sheeting, roof matrix	Asbestos fibers, RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs	The roofing materials were sampled in accordance with the original SAP (DOE-RL 1998).
20	Miscellaneous routine waste	Hg-containing components in manometers, lead associated with incandescent bulbs, PCBs associated with light ballasts, tritium sources for exit signs, sodium bulbs.	Oils, conductor fluids, lead buttons, source units	RCRA metals, Hg (total), PCBs, H-3, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, and Co-60	"Soft" waste and "step-off pad" waste should be included in this waste stream. Generation of this waste stream is ongoing through the completion of this project.
21	Paint	Lead, cadmium, PCBs in paint; radiological contamination from chemical fire, process solution blow-back, and use as fixative	Paint chips	RCRA metals, Pb (total), Cd (total), Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs	Paint samples were collected in the non-process areas, loadout hood and loadout room, viewing room, and stairwell in accordance with the original SAP (DOE-RL 1998).

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
22	Process hood floor (dirt/debris)	Chemical fire	Dirt, debris	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, NO_3^-, and PCBs	Samples of dirt/debris collected during dry cleanup in accordance with the original SAP (DOE-RL 1998). More dirt and debris is being accumulated during D & D.
23	Lubricant/oil	Oil/lubricant components, equipment use	Used equipment oil and grease	PCBs, RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, and Co-60	May need to sample grease on the 233-SA exhaust fans/motors prior to removal.
24	Soil	Chemical fire	Soil from contaminated locations	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, NO_3^- and PCBs	Will need to be obtained prior to or during demolition of the 233-S Facility.
25	Pu recycle tank (L-16)	Chemical process flow, chemical fire	Scale, powder, possible liquid	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60 and NO_3^-, PCBs, and hexone If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	The vessel has been sampled and removed from the 233-S Facility.
26	French drain	Exhaust/air condensate collection	Soil/gravel, possible liquids	RCRA metals, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and NO_3^- If liquids are found, then VOA/SVOA and nitric acid (pH) become COCs.	Will need to be obtained prior to or during demolition of the 233-S Facility.

Table 1-2. Waste Stream[a] in the 233-S Facility and Related COPCs (at the End of Fiscal Year 2001). (6 Pages)

WS #	Waste Stream	Known or Suspected Source of Contamination	Type of Contamination	COPCs	Current Status
27	Anomalies	Chemical process, chemical fire	Solid/liquid	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cm-242, Cs-137, U-238, Co-60, and PCBs Chemical constituents unknown until their purpose and use is evaluated.	Samples collected from the process hood vessels as a result of residual liquid or scale/sludge found within the vessels are: smear from vessel vent line, PMMA panel gasket material, L-4 vessel liquid, L-4 vessel scale/sludge, L-12 vessel liquid, and L-6 vessel scale/sludge.

[a] Represents the worst-case of all of the expected categories of waste streams.
ACM = asbestos-containing material
PCB = polychlorinated biphenyl
PMMA = polymethyl methacrylate
RCRA = *Resource Conservation and Recovery Act of 1976*
SVOA = semi-volatile organic analyte
VOA = volatile organic analyte

1.8.2 Contaminant of Potential Concern Exclusions

Table 1-3 presents a list of the COPCs to be excluded from the investigation. These exclusions are based on physical laws, process knowledge, task focus, or other mitigating factors. Table 1-3 provides the specific rationale for the exclusion of each of the identified COPCs for the associated waste streams.

Table 1-3. Rationale for COPC Exclusions.

Current WS #	COPCs That Are Excluded	Rationale for Exclusion
1, 5, 7, and 26	Cm-242	Cm-242 is a short-lived radionuclide having a half-life of 163 days, which means that the original concentration of Cm-242 would have had to exceed 1.6E+8 Ci/g in order to be greater than 1 pCi/g, currently. This value exceeds the specific activity of Cm-242 by several orders of magnitude.
	NO_3^-	*NO_3^- itself is not a hazardous or dangerous* waste in accordance with Washington State regulations.
	VOA/SVOA	If the LEL is negative and there is no evidence of phase separation in liquids, then VOA/SVOA can be excluded. Process knowledge indicates that there should be no VOA/SVOA present.
3, 9, 15, 16, 22, and 24	Cm-242	Same as above.
	NO_3^-	Same as above.
17, 18, 23	Cm-242	Same as above.
14	Cm-242	Same as above.
	NO_3^-	Same as above.
	DOP	No basis exists for the action levels for waste designation for used DOP.
20	Cm-242	Same as above.
	H-3	H-3 would cause a radiological designation if present in concentrations exceeding 400 pCi/L; however, historical data do not support its designation. It was included for potential exit signs; this type of sign is not present in the 233-S Facility.
27	Cm-242	Same as above.
	Chemical constituents unknown until their purpose and use is evaluated.	Chemical constituent; acetylene tetrabromide (if present) is not regulated.

LEL = lower explosive limit

1.8.3 Final List of Contaminants of Concern

Table 1-4 presents the final list of COCs for each waste stream number to be carried through the remainder of the DQO process.

Table 1-4. Final List of COCs.

Current WS #	Radiological COCs	Chemical COCs
1	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals and PCBs If LEL is negative and there is no evidence of phase separation in liquids, then VOA/SVOA can be excluded. Process knowledge indicates that there should be no VOA/SVOA present.
3, 9	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals
15, 16, 22, and 24	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals and PCBs
5, 7, and 26	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals If LEL is negative and there is no evidence of phase separation in liquids, then VOA/SVOA can be excluded. Process knowledge indicates that there should be no VOA/SVOA present.
14	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals
17	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	Asbestos and PCBs
18	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals, Cr (total), Ni (total), and PCBs
20	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals, Hg (total), and PCBs
23	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	RCRA metals and PCBs
27	Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Np-237, Cs-137, U-238, and Co-60	PCBs, chemical constituents unknown until their purpose and use is evaluated.

1.8.4 Distribution of Contaminants of Concern

The plutonium concentration process, unplanned spills and fires, and discharges resulted in potential contamination of plant components, the plant structure, and soils with the radionuclides, *Resource Conservation and Recovery Act of 1976* (RCRA) metals, nitrates and fluorides. The pipe insulation, lead-based paints, shielding material, roofing material, and use of lubricants resulted in the potential contamination by asbestos fibers, lead, volatile organic compounds, and semi-volatile organic compounds.

The distribution of the COCs in the plant components is predictable (e.g., as vessel residues), localized (e.g., residing inside pipe elbows), and heterogeneous (e.g., not uniformly distributed in a plant component).

1.9 ACTION LEVELS

The action levels that apply to each of the COCs are presented in Table 1-5a with the basis for each action level. The action level is defined as the threshold value that provides the criteria for choosing between alternative actions (AAs).

Table 1-5a. Action Levels for Waste Designation. (2 Pages)

COCs	Action Level	Regulatory Basis
Am-241, Np-237, Pu-238, Pu-239, Pu-240, Pu-242	TRU[b]/GTCC[c]	ERDF waste acceptance criteria (BHI 1998)
Co-60	3.81E+6 pCi/g	Safety analysis for the ERDF (BHI 2001e)
Cs-137	2.67E+7 pCi/g	Safety analysis for the ERDF (BHI 2001e)
Pu-241	GTCC	ERDF waste acceptance criteria (BHI 1998)
U-238	7.5E+3 pCi/g[d]	ERDF waste acceptance criteria (BHI 1998)
Asbestos	1%, by weight	40 CFR 61, Subpart M
Cd (total)[a]	20 mg/kg	WAC 173-303-090(8); 40 CFR 261.24
Cr (total)[a]	100 mg/kg	WAC 173-303-090(8); 40 CFR 261.24
Hexone	100 mg/kg	WAC 173-303-100(5)
Hg (total)[a]	4 mg/kg	WAC 173-303-090(8); 40 CFR 261.24
Ni (total)	220 mg/kg	40 CFR 268.48
Pb (total)[a]	100 mg/kg	WAC 173-303-090(8); 40 CFR 261.24
PCBs	2 mg/kg[e]	WAC 173-303-9904
pH	≤2, ≥12.5	WAC 173-303-090(6); 40 CFR 261.22
TCLP-Ag	5 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP-As	5 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP- Ba	100 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP-Cr	5 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP-Cd	1 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP-Hg	0.2 mg/L	WAC 173-303-090(8); 40 CFR 261.24

Table 1-5a. Action Levels for Waste Designation. (2 Pages)

COCs	Action Level	Regulatory Basis
TCLP-Pb	5 mg/L	WAC 173-303-090(8); 40 CFR 261.24
TCLP-Se	1 mg/L	WAC 173-303-090(8); 40 CFR 261.24

Note: In cases where both TRU and GTCC are listed as action levels, the isotope is subject to both limits and the more limiting of the two will be considered to be the action level.
[a] 20 times the TCLP limit.
[b] TRU waste is defined as radioactive waste containing more than 100 nCi of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years. The term "transuranic" refers to those elements with an atomic number greater than that of uranium (i.e., atomic number >92).
[c] Greater than Class C (GTCC) waste is defined in 10 CFR 61.55, "Licensing Requirements for Land Disposal of Radioactive Waste – Waste Classification."
[d] Assuming the waste density to be 1,600 kg/m^3.
[e] Applies only to electrical-related equipment. Most wastes coming from the 233-S Facility are contaminated with PCBs from other sources, in which case action levels in 40 CFR 761 would apply.
CFR = *Code of Federal Regulations*
WAC = *Washington Administrative Code*

Table 1-5b presents the action levels for criticality safety.

Table 1-5b. Criticality Safety Limits and Criticality Safety Controls for Defense-in-Depth.[a] (2 Pages)

Parameters	Criticality Safety Limits and Criticality Safety Controls	Basis
Pu (total) mass in collection containers, glove bags, removed components, or designated storage arrays outside the process hood after NDA (TSR 5.3.7)	Pu (total) mass (mean + 3σ) ≤450 g together in a single location as measured by NDA	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 4, page 1)
Physical separation between individual collection containers, components and waste packages, which may contain Pu and other accumulations of Pu that are located outside the process hood prior to NDA (TSR 5.3.6)	>24 in.	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 3, page 1)
Physical separation from all other accumulations of Pu of individual collection containers, components, waste packages or designated storage arrays outside the process hood after NDA (TSR 5.3.8)	>24 in.	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 4, page 1)
Pu (total) mass in each 55-gal specification 17H, 17C, UN1A2/X425 and UN1A2/Y400 drum	Pu (total) mass ≤100 g, as measured by NDA	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 3, page 5)

Table 1-5b. Criticality Safety Limits and Criticality Safety Controls for Defense-in-Depth.[a] (2 Pages)

Parameters	Criticality Safety Limits and Criticality Safety Controls	Basis
Pu (total) mass in each 3 ft by 4 ft by 5 ft standard waste box	Pu (total) mass ≤325 g, as measured by NDA	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 3, page 5)
Pu (total) mass in each Connex box	Pu (total) mass ≤450 g, as measured by NDA	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 3, page 5)
Perform NDA of all items out of or associated with the process hood (e.g., piping, vessel sections, and conduit)	>50 mrem/hr (beta/gamma)	*Criticality Safety Program Requirements for 233-S*, 0233S-DB-G0005 (BHI 2001d) (see Attachment 3, page 6) NDA logbook HNF-N-1641 (or successor logbooks)

[a] A revision of BHI (2001d) has been submitted to RL that, when approved, will eliminate all the above TSRs and criticality safety defense-in-depth controls. Following the approval of BHI (2001d), measurements associated with gram quantity of Pu (total) will not be taken.

2.0 PLAN FOR PROJECT TASK ACTION

The *Engineering Evaluation/Cost Analysis for the 233-S Facility* (DOE-RL 1996a) presented four alternative approaches for future facility management and the levels of safety that may be anticipated. Decontamination and/or stabilization of the facility, followed by its demolition and disposal, was selected as the most responsive approach to safety concerns and based on planned land remedial actions. The selection was verified in the March 1997 action memorandum that provides direction to proceed with this non-time critical removal action project (EPA 1997).

The 233-S Facility removal will be performed in a manner that will permit the early disposal of the major fissile material inventory, followed by building cleanup and dismantlement efforts. The early elimination of hazardous substances and conditions will reduce the precautionary measures and the safeguards needed to protect workers and the environment, and will permit the use of standard decommissioning practices. Sampling and analysis will be performed throughout the removal project to provide information for worker safety, protection of the environment, and characterization of various waste streams. This information will be used to dictate the protective controls required for workers involved in specific operations and the preparation of the waste profile summaries for waste disposition at the ERDF. Characterization will be performed in conjunction with planned operations based on accessibility of piping systems within the facility. These systems will only be accessible as D&D operations occur. Throughout the duration of the project, facility conditions will change and/or additional information will become available, which may alter the initial characterization plans.

Removal of the 233-S Facility will be completed when the building and all subsurface structures (to a depth of 0.9 m [3 ft] below grade) have been removed. Contaminated soils down to this level will be excavated and appropriately disposed.

2.1 DECISION MAKERS AND TECHNICAL STAFF

The decision makers for each organization and the technical support staff are listed in Table 2-1. Personnel consulted during the DQO process are also identified.

Table 2-1. Decision Makers and Technical Staff. (2 Pages)

Participant	Responsibility	Participation			
		Aug. 25, 1997	Aug. 27, 1997	Sept. 11, 1997	Sept. 24, 2001
Jeff Bruggeman	DOE	X			
Allan Chaloupka	Project Manager	X		X	X[a]
Julie Atwood	Waste Management/ Transportation				X
Bill Price	Environmental Technologies				X
Scott Thoren	Deputy Task Lead	X	X	X	
George Carter	D&D Engineer	X		X	
Cheryl Volkman	Quality Services	X			
Charlie Blankenship	Field Support	X	X		
Dan Moder	Waste Management				X
Mike Peloquin	Waste Management				X
Les Davenport	Criticality Safety				X
Andy Larson	Nuclear Safety				X
Ryan Johnson	Waste Management	X	X	X	
Randy Jackson	Waste Management	X	X	X	
Greg Borden	Waste Management				X
David Encke	D&D Characterization (Scientist)	X	X	X	X
Rikki Harris	D&D Characterization	X	X	X	
Richard Weiss	Sample Management	X			X
Wendy Thompson	Sampling and Analysis	X			X
Joe Zoric	Environmental Compliance	X	X		
Robert Nielson	Environmental Compliance				X
Mark Kornish	Health Physics Engineer	X		X	
Kevin Funke	Radiological Engineer				X
Grant Ceffalo	Health Physics Engineer				X
Roger Ovink	Environmental Science (Facilitator)	X	X	X	

Table 2-1. Decision Makers and Technical Staff. (2 Pages)

Participant	Responsibility	Participation			
		Aug. 25, 1997	Aug. 27, 1997	Sept. 11, 1997	Sept. 24, 2001
Surajit Amrit	Environmental Technologies (Facilitator)				X
Roy Bauer	Environmental Science				X
David St. John	Sampling and Analysis	X			
Nelson Little	Project Engineer	X			X
Richard Arthur	PNNL NDA			X	
Bruce Gillespie	Canberra NDA				X
Marty Winterose	Canberra NDA				X
Scott Peterson	Environmental Science (Statistician)			X	

[a] Was unavailable at the September 24, 2001, meeting in which the scope of work for the revision of the existing SAP was discussed. Was represented by his project staff.
PNNL = Pacific Northwest National Laboratory

2.2 PROJECT TASK SCOPING AND ISSUES SUMMARY

2.2.1 Data Quality Objective Checklist/Binder

The DQO checklist was prepared to identify the roles and responsibilities of the technical team. The checklist was also prepared to identify additional data needs and personnel responsible for obtaining that data. The DQO binder was prepared by gathering information from historical documents, drawings, radiological surveys, D&D project plans, and personnel interviews. Information from the binder was then used to prepare the DQO scoping report. The scoping report was distributed to the technical team and decision-makers prior to the DQO process.

2.2.2 Scoping Process Issues

The selected approach initially involves removing systems and building features that are known or suspected to contain significant fissile material inventory. Specific waste streams will be identified. However, it is realistic to assume that as the building layers are removed during D&D activities, waste streams that have not been specifically called out will be identified.

2.2.3 Global Issues/Data Quality Objectives Meeting Summary

The initial global issues meeting was held on August 25, 1997, during which the project scoping document, project tasks, participant responsibilities, and scheduled deliverables were discussed. As part of the discussion, the technical team also discussed EPA issues identified during the interview process and identified issues concerning laboratory use, waste stream identification, waste stream disposition, disposal options, and use of NDA equipment.

Another global issues meeting was held on September 24, 2001, during which it was decided that the existing DQO summary report information provided in DOE/RL-97-87, Rev. 0 (DOE-RL 1998) would be revised to clearly establish the proposed use of NDA and to define the NDA criteria for designating waste items generated from the 233-S Facility. In this meeting, it was also decided to provide guidance for the use of radiological survey instruments for obtaining the survey data for radiological control. This data/information is deemed acceptable to support waste management decisions.

2.2.4 Laboratory Issues

The most difficult aspect of the sampling activity is onsite/offsite laboratory acceptance of extremely high alpha-contaminated materials. If necessary, field extraction of highly contaminated samples will be performed to reduce contamination levels prior to shipment.

The characterization team and Sample Management will evaluate the sample volumes, turnaround times, and analysis methods prior to sample collection. All information will be documented on the sampling authorization form (SAF). This will ensure that appropriate laboratory and sample preparation will take place.

Laboratory data will be validated to at least Level C, the minimum level in which quality control (QC) samples are obtained and compared.

2.2.5 Nondestructive Assay

The NDA method, performance requirements, measurement process, and measurement uncertainty for the radiological designation of wastes have been included as a part of the seven-step DQO process in Rev. 1 of DOE/RL-97-87. This is reported in Section 4.0 of this SAP.

2.2.6 Anomalies

It is realistic to assume that as D&D progresses, liquids and residual solids (anomalies) will be found. It was determined that these wastes will be appropriately accumulated for sampling and analysis. Prior to sample collection, sample personnel, D&D Characterization, Waste Management, and Radiological Engineering will evaluate the waste streams to confirm that the sampling approach is appropriate, and that the requested analysis meets the needs of Waste Management to properly identify the radiological and chemical COCs.

2.2.7 Disposal Alternatives

The primary disposal option, as identified in the action memorandum for the 233-S Plutonium Concentration Facility, for each waste stream is the ERDF. Other disposal options (e.g., CWC) will comply with the waste management plan (found in Appendix E of the 233-S removal action report [BHI 2000e]).

2.2.8 233-S Waste Streams/Historical Model

The following information includes waste stream identification, disposal options, and discussion of the historical model for the 233-S Facility. This information will provide the basis for the subsequent steps in the DQO process.

Based on historical information, numerous facility inspections, sampling/analysis results, and detailed radiological surveys, the ERC has identified waste streams within the 233-S Facility. These waste streams will be managed under CERCLA to allow for disposal at ERDF or transportation to the CWC for TRU/GTCC waste.

The historical model for the 233-S Facility was developed based on previous sampling and analysis data, radiological survey data, site and process history, and known sources of contamination types in each waste stream. This information provides acceptable knowledge to support waste characterization and designation. It was used to develop the isotopic ratios for the facility, prepare the ERDF waste profile, and establish acceptable scaling factors used to calculate the radiological inventory in curies per cubic meter (Ci/m^3) per isotope when NDA and radiological surveys are used. Table 1-2 summarizes the historical data, the source of data, COCs for each waste stream, and its status at the end of fiscal year 2001 (FY01).

2.2.9 Environmental Restoration Contractor Waste Management

Prior to disposal, BHI Waste Management will need to ensure proper waste characterization, verification, and designation to satisfy Federal and state applicable or relevant and appropriate requirements and the receiving facilities' waste acceptance criteria.

All waste streams will be certified, radiologically and chemically, through process knowledge and/or approved sampling and analytical methods. This information will be used by Solid Waste Management to prepare waste profile summaries. The required data to prepare waste profile summaries are listed in Table 2-2.

Table 2-2. Required Actions for Waste Designation.

Characterization (ERDF, CWC)	Criteria
1. Determine if the waste is regulated as a listed dangerous waste.[a]	WAC 173-303-080,-081, and -082
2. Determine the applicability of characteristic waste codes: corrosivity, ignitability, reactivity, and toxicity.	WAC 173-303-090(2)-(8)
3. Determine if a waste meets the definition of a toxic dangerous waste (i.e., those wastes with equivalent concentrations of toxic components of 0.001% or more).	WAC 173-303-100(5)
4. Determine if a waste meets the definition of a persistent waste: those wastes that contain a total concentration of halogenated hydrocarbons of 0.01% or more, or a total concentration of polycyclic aromatic hydrocarbons of 1.0% or more.	WAC 173-303-100(6)
5. Determine if the waste is regulated due to its PCB concentrations.	40 CFR 761
6. Determine constituents that may be regulated for land disposal if the waste is designated as dangerous.	WAC 173-303-140; 40 CFR 268
7. Determine the reportable quantities of radiological constituents.	49 CFR 171-173
8. Determine activities of radiological constituents.	Used to support the preparation of waste profiles to compare against the acceptance criteria at ERDF (BHI 1998) or CWC (FH 2001)

[a] No listed waste is expected to be generated during the 233-S Facility D&D Project.

3.0 DATA QUALITY OBJECTIVES SUMMARY REPORT FOR CHEMICAL AND RADIOLOGICAL (WITHOUT USING NONDESTRUCTIVE ASSAY SERVICES) DESIGNATION OF WASTES

3.1 STEP 1 – PROBLEM STATEMENT

The 233-S Facility has been inactive for over 25 years and has no identified future use that would justify a partial cleanup/maintenance approach. The building is radioactively contaminated and has undergone structural deterioration because of exposure to the extreme weather conditions.

Removal of the 233-S Facility will be performed in a sequential progression of operations designed to initially eliminate the most hazardous conditions, followed by a logical course of operations for removal. Sampling and analysis will be performed throughout the removal project to provide information for worker safety, protection of the environment, and identification of various waste streams. This information will be used to dictate the protective controls required for workers involved in D&D operations and to develop the waste profile summaries to support waste disposition decisions.

3.2 STEP 2 – IDENTIFY THE DECISIONS

The list of decisions with potential actions follows. These decisions are focused on waste stream segregation, storage or disposal options, and criteria to meet the storage or disposal options:

1. Determine the waste stream boundaries to optimize sampling and analysis efforts.
2. Determine the nature and extent of contamination of each waste stream, including a determination of whether the waste stream contains dangerous waste, low-level radioactive waste, mixed waste, hazardous waste, TRU waste, or TRU-mixed waste.
3. Determine the storage or disposal options for each waste stream, including whether the waste will be disposed at the ERDF, stored at the CWC or the Transuranic Waste Storage and Assay Facility (TRUSAF), or other EPA-approved storage or disposal site.

3.3 STEP 3 – IDENTIFY INPUTS TO THE DECISIONS

The information required to resolve each decision is listed in Table 3-1, which also lists sources of information needed, and the use of the information in the decision.

Table 3-1. Decisions, Inputs, Source, and Use.

Decision	Input	Source	Use
1. Determine the boundaries of each waste stream to optimize the sample, laboratory, and cost efficiency.	Process knowledge, Table 1-1	Historical documents, facility drawings, radiological surveys, sampling/analytical data	Assess waste stream matrices and COCs
2. Determine if the waste stream contains dangerous waste, low-level radioactive waste, mixed waste, hazardous waste, TRU waste, or TRU-mixed waste.	Process knowledge	Historical data, 1996 final characterization report of the non-process area, radiological surveys	Identify the COCs for requested analysis
	Disposal levels	ERDF and CWC waste acceptance criteria	Compare levels versus sampling results
	Input from sample data	Characterization data	Prepare waste profile for final waste designation, develop isotopic ratios and scaling factors
3. Determine the disposal options for each waste stream. Will the waste be disposed of at the ERDF or CWC?	Inputs from decision #2 above	See decision #2	Use data to assess options, cost, and packaging requirements
	Waste profile	Waste profile summary	Used to determine cumulative total of radionuclides, and concentrations of metals versus disposal criteria

The information needed is listed below:

1. NDA information for criticality evaluations, hot spot identification prior to sample location as reported in Table 3-3. This NDA information is not used for waste designation. NDA measurements for waste designation are addressed in Section 4.0 of this SAP.

2. Radiological surveys (using hand-held instruments) consisting of direct measurements, technical smears, and large area swipes will be used to identify fixed and removable contamination and provide dose-rate information. The radiological surveys will be performed prior to sample collection to identify worst-case radiological concentrations of the waste stream matrices. In-progress surveys used to evaluate changing radiological conditions will also be used to verify and/or update radiological status of the Facility. Surveys will be performed in accordance with BHI-RC-04, Instruction 4.2.

3. Sample collection and laboratory analysis to identify contaminant concentrations. The laboratory data will be used to prepare waste profile summaries that determine waste disposal options.

3.4 STEP 4 – DEFINE THE BOUNDARIES OF THE STUDY

The project boundary for this DQO includes the 233-S Facility internal equipment, components, and building and subsurface structures (to a depth of 0.9 m [3 ft] below grade). Localized contamination found below the 0.9-m (3-ft) level may be removed; however, extensive soil remediation (i.e., locating and removing extensive contamination migration) is not part of this project.

3.5 STEP 5 – DEVELOP DECISION RULES

The decisions were presented in Section 3.2. Decision #1 was made by the technical team, based on process knowledge, waste stream matrix, and waste stream location. Decisions #2 and #3 and associated decision rules (DRs) are listed below.

For decision #2, determine if the waste streams contain dangerous waste, polychlorinated biphenyl (PCB) waste, low-level radioactive waste, mixed waste, hazardous waste, TRU waste, or TRU-mixed waste.

- If the sample obtained from waste streams exceeds the dangerous waste criteria (*Washington Administrative Code* [WAC] 173-303-080, -081), then the waste must be treated as dangerous waste.
- If the sample obtained from waste streams exceeds the PCB waste criteria (40 *Code of Federal Regulations* [CFR] 761), then the waste must be treated as PCB waste.
- If contamination concentrations exceed the dangerous waste criteria (WAC 173-303-090[2]-[8]), then the waste must be treated as hazardous waste.
- If radiological contamination is present, then the material is radioactive.
- If radiological contamination is present at levels not exceeding the TRU waste criteria (DOE O 435.1) and the material contains PCB wastes, then the material is radioactive and must be treated as PCB LLW.
- If the waste meets the mixed waste criteria as defined by DOE O 435.1 and WAC 173-303, then the material must be treated as mixed waste.
- If contamination concentrations exceed the TRU waste criteria (DOE O 435.1), then the material must be treated as TRU waste.
- If contamination concentrations exceed the TRU waste criteria (DOE O 435.1) and the material contains PCB wastes, then the material must be treated as PCB-TRU waste.
- If contamination concentrations exceed the dangerous waste criteria (WAC 173-303) and the TRU waste criteria (DOE O 435.1), then the material must be treated as TRU-mixed waste.

- If contamination concentrations exceed the GTCC waste criteria (10 CFR 61.55) and are less than the TRU waste criteria (DOE O 435.1), the material must be treated as GTCC waste.

For decision #3, determine the storage/disposal options for each waste stream. Will the waste be disposed of by transportation to ERDF, CWC, or TRUSAF?

- If the waste stream profile does not comply with the ERDF waste acceptance criteria, then the waste will be stored at the CWC or TRUSAF.

3.5.1 Parameters of Interest

3.5.1.1 Facility Structure and Internal Components. Process knowledge, NDA information, and radiological surveys indicate that the waste matrices are radiologically contaminated. There are worst-case (or "hot spot") locations in the pipe systems, ducts, and areas of the facility structure that have a higher potential for radioactive material buildup. Process knowledge also shows that these hot spots are correlated with a buildup of hazardous chemical concentrations as well. A sampling design (based on professional judgment) and worst-case (authoritative) sampling will be used to determine the maximum levels of radiological contamination. The parameter of interest will be a single maximum analytical value for every constituent in each waste stream that will be compared with the waste acceptance criteria decision levels. This design is described in further detail in DQO Step 7 (see Section 3.7).

3.5.1.2 Soil Excavation. Removal of the 233-S Facility and its systems will be completed to a depth of 0.9 m (3 ft) below grade. Soil characterization of the excavated area will be performed by detailed radiation surveys and the analysis of representative soil samples, as specified in *Soils and Solid Media* (EPA 1989) and *Guidance on Sampling and Data Analysis Methods* (Ecology 1995). These efforts will involve establishing a grid system on the area and performing radiological surveys. These surveys will verify the remaining conditions at the conclusion of the removal activities and establish radiological postings and boundaries as required by BHI-RC-04, Instruction 6.2, "Posting Radiological Areas." The surveys are not intended to support any decision concerning unconditional release of the site as required by DOE Order 5400.5, *Radiation Protection of the Public and the Environment*. This information should be provided to Waste Information Data System to update the status of this waste site. If the remaining soil is contaminated, with DOE/EPA concurrence, further remediation will become the responsibility of a future remedial action. A cap of clean borrow soil will be placed over the involved area, and routine surveillance activities will be initiated.

3.5.2 Decision Levels

The COCs and waste decision criteria for waste designation are summarized in Table 3-2. The COCs in this table are based on process knowledge of known contaminants in the identified waste streams. The COCs also represent analyses needed to identify unknown waste streams that may be discovered during D&D activities. The waste decision criteria identified in Table 3-2 are based on the required actions for waste designation listed in Table 2-2.

3.6 STEP 6 – SPECIFY LIMITS ON DECISION ERRORS

Because a statistical sampling approach is not feasible or deemed necessary for the 233-S Facility's waste streams, professional judgment, and a worst-case (maximum COC concentration), authoritative sample design will be applied.

3.7 STEP 7 – OPTIMIZE THE DESIGN

3.7.1 Analysis Criteria

The laboratory analyses, methods, waste decision criteria, and laboratory detection limits are listed in Table 3-2. The table indicates the laboratory analysis for identified waste streams, as well as anomalies that may be found. Process knowledge will be evaluated by ERC Waste Management/Transportation prior to sampling activities to eliminate or add analyses, if appropriate.

The radiological surveys will be performed prior to sample collection to identify areas of worst-case radiological contamination for sampling of the waste stream matrices and to verify/update process knowledge, as necessary. Surveys will be performed in accordance with BHI-RC-04, Instruction 4.2. A combination of static and scan measurements will be performed to evaluate contamination. Technical smears and large area wipe samples will be collected, as needed, to evaluate removable contamination. The results of the field surveys will be documented in accordance with BHI-RC-04, Instruction 4.2 and will be used as an aid in designating waste streams.

3.7.2 Sample Optimization

The sampling design for the 233-S Facility structure and internal equipment is based on a "worst-case" (maximum COC concentration) sampling approach that identifies accessible locations where sufficient information for safety considerations and waste designation can be applied.

Table 3-2 describes the laboratory detection limits and sample volumes needed for each analysis requested. The sample volumes are separated into maximum volumes for full protocol analysis and minimum volumes for quick-turnaround data. Table 3-3 provides the sample strategy and rationale for each waste stream, and suspect matrices for each waste stream. Previous sampling activities at the 233-S Facility, process knowledge, and as low as reasonably achievable (ALARA) information indicate that under most circumstances maximum volume collection may not be achieved. Each sample location will be evaluated on a case-by-case basis to determine if full protocol will be used or if minimum volume collection will be used for quick-turnaround data.

Table 3-2. Laboratory Detection Limits for the COCs at 233-S Facility. (2 Pages)

COCs	Analytical Callout	EPA Method	Laboratory Accuracy and Precision[a]	Analytical Technique	Commercial Laboratory			
					Detection Limits[b]		Volume Requirements[b]	
					Solid[c]	*Liquid*[c]	*Solid*[d]	*Liquid*[d]
Pu-238, Pu-239/240, Pu-241, Pu-242	Pu isotopic	Laboratory-specific	a	AEA	1 to 20	1 to 20	25 to 4	600 to 50
Am-241	Am/Cm isotopic	Laboratory-specific	a	AEA	1 to 20	1 to 20	25 to 4	600 to 50
Np-237	Np-237	Laboratory-specific	a	AEA	1 to 20	1 to 20	25 to 4	600 to 50
Co-60[e]	GEA	Laboratory-specific	a	GEA	0.1 to 1	25 to 100	1500 to 50	1,500 to 50
Cs-137	GEA	Laboratory-specific	a	GEA	0.1 to 1	15 to 100	1,500 to 50	1,500 to 50
Gross alpha	Gross alpha	Laboratory-specific	a	Proportional counting	10 to 25	3 to 7	2 to 0.5	600 to 150
Gross beta	Gross beta	Laboratory-specific	a	Proportional counting	15 to 30	4 to 8	2 to 0.5	600 to 150
Acids	pH	SW-846, Method 9040/9041A	a	Electrode/paper	0.1 to 0.1	0.1 to 0.1	10 to 3	100 to 25
PCBs	PCBs	SW-846, Method 8082	a	GC	0.05 to 10	0.5 to 100	120 to 1	2,000 to 20
Chromate, SS steel corrosion-chromium	Cr (total)	SW-846, Method 6010A	a	ICP	0.5 to 5	3 to 20	15 to 2	500 to 150
SS steel corrosion - nickel	Ni (total)	SW-846, Method 6010A	a	ICP	4 to 10	20 to 100	15 to 2	500 to 150
Nitrates	NO_3^-	EPA Method 300.0	a	IC	0.1 to 5	10 to 50	40 to 5	300 to 50
TCLP	TCLP – Pb	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	250 to 500	300 to 25	500 to 150
TCLP	TCLP – Cr	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	3 to 20	300 to 25	500 to 150
TCLP	TCLP – Cd	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	5 to 10	300 to 25	500 to 150

Table 3-2. Laboratory Detection Limits for the COCs at 233-S Facility. (2 Pages)

COCs	Analytical Callout	EPA Method	Laboratory Accuracy and Precision[a]	Analytical Technique	Commercial Laboratory: Detection Limits[b]		Commercial Laboratory: Volume Requirements[b]	
					Solid[c]	Liquid[c]	Solid[d]	Liquid[d]
TCLP	TCLP – Ag	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	5 to 10	300 to 25	500 to 150
TCLP	TCLP – As	SW-846, Method 1311/6010A	a	Extraction –ICP	Extract[f]	5 to 10	300 to 25	500 to 150
TCLP	TCLP – Se	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	5 to 10	300 to 25	500 to 150
TCLP	TCLP – Ba	SW-846, Method 1311/6010A	a	Extraction – ICP	Extract[f]	5 to 10	300 to 25	500 to 150
TCLP	TCLP – Hg	SW-846, Method 1311/7470	a	Extraction – CVAA	Extract[f]	0.5 to 2	300 to 25	500 to 150
Asbestos	Asbestos	N/A	a	Microscopy	N/A	N/A	N/A	N/A

[a] Precision and accuracy requirements for both commercial and onsite laboratories are established prior to testing. The basis for measurement accuracy and precision is specified in Vol. 4, Section 7.0 of the *Hanford Analytical Services Quality Assurance Requirements Document* (DOE-RL 1996b).

[b] First value is for "full protocol;" second value is for rapid turnaround or reduced volume analysis. Full protocol detection limits require the larger volume shown. Detection limits are based on optimal conditions. Sample-specific matrix effects or interference may raise the values shown. Detection limits are minimum detection activities for radionuclides and minimum detectable concentrations for chemicals.

[c] Values in pCi/g or mg/kg for solids, and pCi/L or µg/L for liquids.

[d] Values in g for solids or mL for liquids. Radionuclide analyses and metals analyses volumes may be combined to reduce total volume needed.

[e] These radionuclides are not considered as COCs for the 233-S Facility. They are used as flags for potential cross-contamination from REDOX, 200 West tank farms, or the 222-S Laboratories.

[f] TCLP values are reported as liquid extract concentrations for solid samples and bulk liquid concentrations for liquid samples.

AEA = alpha energy analysis
CVAA = cold vapor atomic absorption
GC = gas chromatograph
GEA = gamma energy analysis
ICP = inductively coupled plasma
TCLP = toxicity characteristic leaching procedure

Table 3-3. 233-S Facility Sampling Strategy for Waste Streams (at the End of Fiscal Year 2001).[a] (3 Pages)

WS #	Waste Stream	Waste Stream Description	Sample Strategy	Sample Method
1	Process Drains	Two process drains connecting to the 202-S return line	1. Visual inspection of drains to confirm if sample matrix is liquid or residual sediment. 2. Because of small sample volume, the sample matrix will be collected from both drains and combined into one composite sample.	Liquids (if present) will be collected from each drain using pipettes with thumb vacuum control, peristaltic pump. Residual solids (if present) will be collected by scraping the solids from the drain.
3	Elemental lead	Located in the process hood area	1. Perform visual inspection. 2. If necessary, collect swipes for radiological information.	Technical smears will be obtained for radiological screening.
5	Np concentrator (L-2 vessel)	7 in. by 23 ft, 44 in. wide overall; 10 ft of Raschig ring packing; 32 turns of 0.75-in. Schedule 40 coil	1. Perform visual inspection to confirm if sample matrix is liquid or residual solid. 2. One sample will be collected from each concentrator. These will not be composite samples.	Liquids (if present) will be collected from each concentrator using pipettes with thumb vacuum control, peristaltic pump. Residual solids (if present) will be collected by scraping the solids from the concentrators.
7	Np concentrator/ condenser (L-8 vessel)	7 in. by 4.5 ft, 20 in. wide overall; 27 turns of 0.75-in. Schedule 40 coil		
9	Process pipe trench (concrete structure)	23 ft 6 in. long, 4 ft 8 in. wide, 3 ft 4 in. deep	1. NDA information and/or radiological surveys will be used to determine worst locations for contamination concentrations. 2. Sampling of the concrete will occur in succession with scabbling activities.	Sample will be collected from the scabble debris using scoop or appropriate sample tool.
14	HEPA filters	8 (24 in. by 24 in. by 11.5 in.) in CWS, 18 (24 in. by 24 in.) in 233-SA	1. Radiological surveys will be conducted to identify worst-case contamination. 2. One core sample will be collected from each filter in the CWS and combined into one composite. One sample will be collected from each filter in the 233-SA and combined into one composite. This will allow for sufficient sample volume as well as a representative sample of the filter system.	Core samples will be collected using a coring tool.

Table 3-3. 233-S Facility Sampling Strategy for Waste Streams (at the End of Fiscal Year 2001).[a] (3 Pages)

WS #	Waste Stream	Waste Stream Description	Sample Strategy	Sample Method
15	Asphalt	N/A	1. NDA information and/or radiological surveys will be used to determine worst locations for contamination. 2. Asphalt samples will be collected from worst-case locations.	Asphalt samples will be collected using saws or drills. Different drills or bits will be used for each boundary. Tools will be field decontaminated between samples.
16	Concrete floors/walls	N/A	1. NDA information and/or radiological surveys will be used to determine worst locations for contamination. 2. Concrete samples will be collected from worst-case locations.	Samples will be collected using saws or drills. Different blades or bits will be used for each boundary. Tools will be wiped clean and surveyed between samples.
17	Asbestos-containing material	N/A	1. AHERA asbestos inspector will perform good faith inspection to determine which material is suspect. 2. Radiological surveys will also be conducted prior to sampling. 3. Samples will be obtained in accordance with simplified sampling scheme for friable surface materials (EPA 1985).	Samples will be collected using chisel, hammer, and pliers.
18	Wood/sheetrock	N/A	1. NDA information and/or radiological surveys will be used to determine worst locations for contamination. 2. Samples will be collected from worst-case locations.	Samples will be collected using saws or drills, depending on the material.
20	Miscellaneous routine waste	Includes light ballasts, smoke detectors, and fluorescent bulbs	1. These are all well-established waste streams (no sampling required).	None.
22	Process hood floor (dirt/debris)	Small pile of debris	1. Perform visual inspection to confirm if sample volume is adequate. 2. NDA information and/or radiological surveys will be used to determine worst-case locations for contamination. 3. If the sample matrix is not of criticality concern, the dirt and debris will be mixed and one sample will be obtained.	Sample will be collected using spoons, scoopulas, or other appropriate method.

Table 3-3. 233-S Facility Sampling Strategy for Waste Streams (at the End of Fiscal Year 2001).[a] (3 Pages)

WS #	Waste Stream	Waste Stream Description	Sample Strategy	Sample Method
23	Lubricant/oil	Found in equipment	1. Lubricant reservoirs in all equipment will be visually inspected. If found, oils will be collected into a single composite sample container.	Samples will be obtained using pipettes or bulb droppers.
24	Soil	N/A	1. Field determination will be used to decide if NDA information and/or radiological surveys will be needed for criticality evaluation prior to sampling. Information will also be used to determine worst-case contamination. 2. Samples from each excavated soil boundary will be collected and combined into one composite. Samples will be collected from worst-case locations.	Samples will be obtained using core or auger equipment. Sample matrix will be combined into one collection container and transferred into sample jars using spoon, scoopulas, or other appropriate method.
26	French drain	Approximately 24 in. diameter, 5 ft deep	1. Perform visual inspection. 2. NDA information and/or radiological surveys will be used to determine contamination concentrations for criticality evaluation prior to sampling. 3. Three samples will be collected and combined into one composite. Based on the diameter of the french drain, three samples should represent data results.	Samples will be obtained using core sampler or auger. Samples will be combined in one collection jar and transferred into sample containers using spoons, scoopulas, or other appropriate methods.
27	Anomalies (liquid/solid)	N/A	1. Unknown liquids and solids will be accumulated in appropriate containers and evaluated to determine sampling and analysis requirements for waste designation.	Liquids (if present) will be collected using pipettes with thumb vacuum pressure. Peristaltic pumps will be used if volume is large enough. Solids (if present) will be scraped, shoveled, and scooped into collection jars for analysis.

[a] Eleven waste streams (2, 4, 6, 8, 10, 11, 12, 13, 19, 21, and 25) that were originally listed in Rev. 0 of DOE/RL-97-87 (DOE-RL 1998) have been designated with characterization data prior to the end of FY01.

AHERA = *Asbestos Hazard Emergency Response Act*

N/A = not applicable

4.0 DATA QUALITY OBJECTIVES SUMMARY REPORT FOR USE OF NONDESTRUCTIVE ASSAY SERVICES IN RADIOLOGICAL DESIGNATION OF WASTES

4.1 STEP 1 – PROBLEM STATEMENT

In order to designate the radiological component of the waste items originating from D&D of the 233-S Facility using NDA measurements, the measurement parameters and the total measurement uncertainty (TMU) need to be established.

4.2 STEP 2 – IDENTIFY THE DECISIONS, AND STEP 3 – IDENTIFY INPUTS TO THE DECISIONS

This step defines the principal study questions (PSQs) that need to be resolved to address the problem previously identified in Section 4.1 and the AAs that would result from the resolution of the PSQs. The PSQs and AAs are then combined into decision statements (DSs) that express a choice among AAs. Table 4-1 presents the task-specific PSQs, AAs, and resulting DSs. This table also provides a qualitative assessment of the severity of the consequences of taking an AA if it is incorrect. This assessment takes into consideration human health and the environment (i.e., flora/fauna) and political, economic, and legal ramifications. The severity of the consequences is expressed as low, moderate, or severe.

Table 4-1. Summary of DQO Step 2 Information. (2 Pages)

PSQ-AA#	Alternative Action	Description of Consequences of Implementing the Wrong Alternative Action	Severity of Consequences (Low/Moderate/Severe)
PSQ #1 – Does the radiological activity of the waste item meet the definition of TRU/GTCC waste?			
1-1	No. Can dispose waste at ERDF.	Negative impact on human health and environment. Imposition of fine/penalty. Loss of professional credibility.	Severe
1-2	Yes. Can transport waste to CWC for storage.	Negative financial impact to project's waste disposal budget. Loss of professional credibility.	Moderate
DS #1 – Determine if the radiological activity of the waste meets the definition of TRU/GTCC waste for disposal at CWC; otherwise dispose at ERDF.			

Table 4-1. Summary of DQO Step 2 Information. (2 Pages)

PSQ-AA#	Alternative Action	Description of Consequences of Implementing the Wrong Alternative Action	Severity of Consequences (Low/Moderate/Severe)
PSQ #2[a] – Does the plutonium mass in the waste item in any single location meets the criticality safety limit [(mean +3 σ) ≤ 0.5 g Pu (total)]?			
2-1	Yes. Allow storage of the waste item in any radiological material storage area (RMSA).	Negative impact on human health and environment. Imposition of fine/penalty. Loss of professional credibility.	Severe
2-2	No. Log the plutonium mass in the waste item and its radioactivity into the fissile inventory and place it in a storage array.	Negative financial impact to project's waste disposal budget. Loss of professional credibility.	Moderate
DS# 2 - Determine if the plutonium mass in a waste item in any single location meets the criticality safety limit [(mean +3 σ) < 0.5 g Pu (total)] to allow its storage in any RMSA; otherwise log the waste item and its radioactivity into the fissile inventory and place it in a storage array.			

[a] A revision of BHI (2001d) has been submitted to RL that, when approved, will eliminate all the above TSRs and criticality safety defense-in-depth controls. Following the approval of BHI (2001d), measurements associated with gram quantity of Pu (total) will not be taken.

Table 4-2 specifies the information (data) required to resolve each of the DSs identified in Table 4-1 and identifies whether the data already exist. For the existing data, the source references for the data have been provided with a qualitative assessment as to whether or not the data are of sufficient quality to resolve the corresponding DS. The qualitative assessment of the existing data was based on the evaluation of the corresponding QC data (e.g., spikes, duplicates, and blanks), detection limits, data collection methods, etc.

Table 4-2. Required Information to Resolve the Decision Statements.

DS #	Required Data	Do Data Exist? (Y/N)	Sufficient Quality? (Y/N)	Additional Information Required? (Y/N)
1 and 2	For NDA data for potentially TRU/GTCC items: • Weight of the item • Dimension/geometry of the item • Material of construction of item • The Pu isotopic ratio and/or other nuclide scaling factors for the nonmeasurable radionuclides • High-resolution gamma analysis of the item	N	N/A	Y

[a] An item could be a plant component, bag of waste, box of waste, etc.
N/A = not applicable

4.2.1 Computational and Survey/Analytical Methods

Table 4-3 identifies the DSs where existing data do not exist or are of insufficient quality to resolve the DSs. For these DSs, Table 4-3 presents computational and/or surveying/sampling methods that could be used to supplement sampling and laboratory analytical data to resolve the DSs.

Table 4-3. Information Required to Resolve the Decision Statements.[a]

DS #	Informational Need	Required Data	Computational Methods	Survey/Analytical Methods
1	Radiological activity in wastes (i.e., plant components, "soft" wastes and "step-off pad" wastes)	Verification of process knowledge	Radiological Engineering calculations used to develop isotopic ratios and scaling factors.	Surveys conducted in accordance with BHI-RC-04, Instruction 4.2. Field instruments will be operated in accordance with BHI-RC-05, Instruction 2.1.
		Radiological activity (in pCi/g)		NDA – high-resolution germanium gamma spectroscopy
2	Plutonium quantity in a TRU/GTCC wastes	Radiological quantity (in g)	N/A	NDA – high-resolution germanium gamma spectroscopy

[a] This information is used to supplement the existing sampling and laboratory data that has established the isotopic ratios.
N/A = not applicable

As reported in Table 4-3, the survey/analytical methods are as follows:

- Portable radiological survey using scintillation detector, ion chamber, and Geiger-Mueller counter – This technique will be used to identify total and removable radiological contamination levels measured in disintegrations per minute/100 cm^2 (dpm/100 cm^2), and for providing dose rate information (in mrem/hr).

- NDA using high-resolution germanium gamma spectroscopy – This technique is used to detect gamma radiation emitted from the contamination on the items being characterized. The gamma radiation is accumulated into a spectrum and analyzed based on gamma-ray energy to identify the radionuclides present in the item.

For radiological surveys performed in accordance with BHI-RC-04, Instruction 4.2 and BHI-RC-05, Instruction 2.1, the instrumentation used is to determine gross beta-gamma contamination and gross alpha contamination. Isotopic analysis is provided using NDA or laboratory analytical techniques (described in Table 3-2).

Decision levels for the radiological surveys are based on the requirements for implementation of the ERC radiological control program and are discussed in the following BHI-RC-04 procedures:

- Instruction 4.2, "Radiological Surveys" (used to locate and quantify the presence of radioactive contamination and verify process knowledge for waste)

- Instruction 6.1, "Radioactive Material Labeling and Packaging" (used to ensure sufficient labeling and packaging of radioactive material/waste as it is generated and while in storage)

- Instruction 6.2, "Posting Radiological Control Areas" (establishes radiological postings and boundaries during D&D activities).

As noted as in the introduction (Section 1.0), the survey to release items and equipment in accordance with BHI-RC-04, Instruction 4.4 is not within the scope of this SAP and is performed required by the procedure to demonstrate release to the applicable decision levels stated in the procedure.

4.2.2 NDA Analytical Performance Requirements

Table 4-4 defines the analytical performance requirements for the data needed to resolve each of the DSs. These performance requirements include the minimum detectable activity (MDA) and precision and accuracy requirements for selected radionuclides.

The MDA is a function of the size, and density of the waste item, the distance of the detector from the waste item, the detector count time, etc. The MDA can be directly calculated by the assay software for each measurement. If measured values are not reported, the NDA analyst will use the calculated MDA result in lieu of the measured value to determine the upper bound for the item activity.

Table 4-4. NDA Analytical Performance Requirements for NDA Measurements.

DS #	Measurable Radionuclides[a]	Survey/ Analytical Methods	Action Level	MDA[b] (Expected)
1	Am-241	NDA	TRU/GTCC	≈20 nCi/g
	Pu-239			≈30 nCi/g
	Np-237			$\approx 10^{-2}$ nCi/g
	Cs-137			$\approx 10^{-1}$ nCi/g
	Co-60			$\approx 10^{-1}$ nCi/g
2	Pu-239	NDA	Pu (total) (mean +3 σ) ≥ 0.5 g[c]	≈0.2 to 0.5 g

[a] Measurable radionuclides are those radionuclides that will be directly measured by NDA. Using the concentration values of these measurable radionuclides and the plutonium isotopic ratio the concentration of the remaining COCs; Pu-238, Pu-240, Pu-241, and Pu-242 can be calculated.
[b] MDA is the lowest possible concentration or mass of a radionuclide that can be reliably measured by NDA method.
[c] The lowest mass of Pu (total) that must be recorded on continuous fissionable material inventory sheets once measured.

To obtain a reasonable quantification of the radionuclide concentrations, an efficiency calibration must be developed to account for the item geometry and the attenuators and matrix elements within the item[1] that will attenuate or absorb the gamma radiation. This is accomplished through a commercial modeling program. The analyst uses the information concerning the item weight, dimensions, material of construction, and expected source distributions to model the item to be assayed. The program generates an efficiency calibration curve for the item. The efficiency curve is applied to the spectrum generated by the germanium detector to provide a quantitative value for the detected radionuclides or a MDA for radionuclides that are not detected. The efficiency calibration curve produced from the modeling accounts for the effects from the attenuation, geometric correction factors, and source distributions.

Not all of the radionuclides of interest that are present in the waste generated from D&D of the 233-S Facility can be directly measured through gamma spectroscopy; therefore, isotopic ratios or scaling factors must be provided for the nondetectable nuclides. These can be obtained through destructive laboratory sample analysis or for the plutonium isotopes can be obtained through specialized plutonium isotopic analysis software. The plutonium isotopic analysis can only be obtained for items with significant (typically gram) quantities of plutonium. The normal calculation for the plutonium content is based on a measured plutonium-239 quantity multiplied by the isotopic ratios for the other plutonium isotopes. Table 4-5 shows the isotopic ratios that will be used to calculate activities for the non-measurable isotopes. This information is based on the waste profile for the 233-S Facility (Profile #233S001, Rev. 3) and is deemed to be the

[1] An item to be assayed could be a plant component, "soft wastes" in a plastic bag or a box, a drum full of pipes, etc.

applicable and appropriate isotopic ratio to be used to determine the concentration of non-measurable isotopes for all the waste items generated from the 233-S Facility.

Table 4-5. Isotopic Ratios.[a]

Radiological Isotope	% by Radioactivity
Np-237	0.00005
Pu-238	7.7
Pu-239	19.0
Pu-240	9.9
Pu-241	37.0
Pu-242	0.01
Am-241	26.3

[a] Waste profile for 233-S Facility - Profile #233S001, Rev. 3, dated October 1, 2001. If future data/information on the 233-S Facility justifies the use of different isotopic ratios, then the above table will be revised to incorporate it. Revision of the table values will not warrant the revision of this SAP.

Classification of the waste as TRU and final values for transportation to CWC requires reporting of all of the measured and calculated nuclide activities.

To calculate the "gram" quantity for the item, the quantified value for plutonium-239 must be converted to gram values by multiplying the nuclide activity by the specific activity (6.204E-2 Ci/g) for plutonium-239. The gram quantity for plutonium-239 is the only nuclide required because plutonium-239 and plutonium-241 are the only fissile nuclides expected to be found in the 233-S Facility waste streams. Reporting of the gram quantity must include the addition of the total measurement uncertainty as calculated at the 3σ level.

4.2.2.1 Detector Resolution. The detector used for the measurements must have adequate resolution to differentiate between gamma ray energies as close as 4 keV apart.

4.2.2.2 Dynamic Measurement Range. The detection system must have the capability to measure items with concentrations ranging from the MDA values listed in Table 4-4 up to 1.0E+6 nCi/g. This may be accomplished through variations in the item to detector distance or through the application of filters that preferentially attenuate the low-energy gamma radiation.

4.3 STEP 4 – DEFINE THE BOUNDARIES OF THE STUDY

The objective of DQO Step 4 is to identify the population of interest, define the spatial and temporal boundaries that apply to each DS, define the scale of decision making, and identify practical constraints that must be considered in the sampling design. Completing this step helps to ensure that the data collected will accurately reflect the true condition of the site being investigated.

4.3.1 Population of Interest

Prior to defining the spatial and temporal boundaries of the site under investigation, it is first necessary to define the populations of interest that apply to each DS. The intent of Table 4-6 is to define the attributes of each population of interest by stating them in a way that makes the focus of the study unambiguous.

Table 4-6. Characteristics that Define the Population of Interest.

DS #	Population of Interest	Unit Measurement Size	Total Number of Potential Measurement Units Within the Population
1 and 2	All waste items representing WS #s 1, 3, 5, 7, 9, 14, 15, 16, 17, 18, 20, 22, 23, 24, 26, and 27 of the 233-S Facility. (see Table 1-4)	The NDA system can measure large waste items; however, multiple measurements will be required.	Unknown

4.3.2 Geographic Boundaries

Table 4-7 identifies the geographic boundaries for each DS. Identifying the boundaries of the study area ensures that the investigation will not expand beyond the original scope of the task.

Table 4-7. Geographic Boundaries of the Investigation.

DS #	Geographic Boundaries of the Investigation
1 and 2	233-S Facility

4.3.3 Zones with Homogeneous Characteristics

Table 4-8 defines the zones within the site that have relatively homogeneous characteristics. These zones are identified by using existing information to segregate the elements of the population into subsets that exhibit relatively homogeneous characteristics (e.g., types of contaminants). Dividing the 233-S Facility into homogeneous zones reduces the overall complexity of the problem by separating the site into more manageable pieces.

Table 4-8. Zones with Homogeneous Characteristics.

DS #	Population of Interest	Zone (refer to Figure 1-5)	Homogeneous Characteristic Logic
1 and 2	All waste items representing WS #s 1, 3, 5, 7, 9, 14, 15, 16, 17, 18, 20, 22, 23, 24, 26, and 27 of the 233-S Facility. (see Table 1-4)	Process hood, viewing room, stairwell	Facility history and process knowledge
		PR can storage room, pipe gallery, equipment room, toilet, SWP change room, electrical cubicle, loading dock, enclosure	
		233-SA Exhaust Filter Building, various airlocks	

4.3.4 Temporal Boundaries

Table 4-9 identifies the temporal boundaries that may apply to each DS. Temporal boundaries refer to the timeframe over which each DS applies (e.g., number of years) and when (e.g., season, time of day, and weather conditions) data should optimally be collected to resolve each DS.

Table 4-9. Temporal Boundaries of its Investigation.

DS#	Timeframe	When to Collect Data
1 and 2	FY02 through the end of the ERC contract	No restrictions if the radiological survey is carried out indoors.

FY = fiscal year

4.3.5 Scale of Decision Making

Table 4-10 documents the scale of decision making for each DS.

Table 4-10. Scale of Decision Making.

DS #	Population of Interest	Geographic Boundary	Temporal Boundary		Scale of Decision
			Timeframe	When to Collect Data	
1 and 2	All waste items representing WS #s 1, 3, 5, 7, 9, 14, 15, 16, 17, 18, 20, 22, 23, 24, 26, and 27 of the 233-S Facility (see Table 1-4)	233-S Facility	FY02 through the end of the ERC contract	No restrictions if the radiological survey is carried out indoors.	The area/rooms as depicted in the Figure 1-5.

4.3.6 Practical Constraints

Table 4-11 identifies practical constraints that may influence data collection efforts (e.g., physical barriers, difficult sample matrices, and high radiation areas).

Table 4-11. Practical Constraints on Data Collection.

• Criticality safety concerns (see Table 1-5b for the criticality safety limits)[a].
• Working in high radiation areas (≥100 mR/hr at 30-cm distance).

[a] A revision of BHI (2001d) has been submitted to RL that, when approved, will eliminate all the above TSRs and criticality safety defense-in-depth controls. Following the approval of BHI (2001d), measurements associated with gram quantity of Pu (total) will not be taken.

4.4 STEP 5 – DEVELOP DECISION RULES

The purpose of DQO Step 5 is to define the statistical parameters of interest (e.g., mean or median) that will be used for comparison against the action levels. In DQO Step 5, the DRs (i.e., "IF…THEN…" statements) are developed for each DS. The DRs typically incorporate the parameter of interest, the scale of decision making (from DQO Step 4), the action level (from DQO Step 1), and the AAs (from DQO Step 2) that would result from resolution of the DS.

4.4.1 Statistical Parameter of Interest

Table 4-12 presents the statistical parameter of interest for each DS.

Table 4-12. Statistical Parameter of Interest.

DS #	Decision Statement	Statistical Parameter of Interest
1	Determine if the radiological activity of the waste meets the definition of TRU/GTCC waste to cause its transportation to CWC for storage; otherwise dispose at ERDF.	Average (of at least two measurements)[a]
2[b]	Determine if the plutonium mass in a waste item in any single location meets the criticality safety limit [(mean +3 σ) < 0.5 g Pu (total)] to allow its storage in any RMSA; otherwise log the waste item and its radioactivity into the fissile inventory and place it in a storage array.	Average (of at least two measurements)[a]

[a] One continuous measurement of a rotated item is also defined as an average measurement.
[b] A revision of BHI (2001d) has been submitted to RL that, when approved, will eliminate all the above TSRs and criticality safety defense-in-depth controls. Following the approval of BHI (2001d), measurements associated with gram quantity of Pu (total) will not be taken.

4.4.2 Decision Rules

Table 4-13 presents the DRs that correspond to each of the DSs identified in Table 4-1.

Table 4-13. Decision Rules.

DS #	DR #	Decision Rule
1	1	If the radiological activity of the waste item as determined by the average NDA measurement meets the definition of TRU/GTCC waste then transport the waste item at CWC or storage. If the radiological activity of the waste item as determined by the average NDA measurement does not meet the definition of TRU/GTCC waste then dispose the waste item at ERDF.
2[a]	2[a]	If the plutonium mass in a waste item(s) together in any single location as determined by the average NDA measurement is below the criticality safety limits then allow its storage in any RMSA. If the plutonium mass in a waste item(s) together in any single location as determined by the average NDA measurement is above the criticality safety limits then log the waste item and its radioactivity into the fissile inventory and place it in a storage array.

[a] A revision of BHI (2001d) has been submitted to RL that, when approved, will eliminate all the above TSRs and criticality safety defense-in-depth controls. Following the approval of BHI (2001d), measurements associated with gram quantity of Pu (total) will not be taken.

4.5 STEP 6 – SPECIFY LIMITS ON DECISION ERRORS

4.5.1 Total Measurement Uncertainty for Nondestructive Assay Method

There are several components that are part of the overall total measurement uncertainty of typical measurements used by the NDA method. The most significant of these components are as follows:

- Measurement statistics and activity ranges
- Source location and uniformity
- Matrix, absorber, geometry uncertainties, and attenuation
- Isotopic ratios and scaling factors.

All of these uncertainties will be considered random for the purposes of this analysis and therefore will be summed in quadrature to obtain a final total measurement uncertainty.

In general, this section is an attempt to provide semi-quantitative uncertainties to qualitative evaluations. It is not intended to be rigorous, as the discussion simply provides the indications and categorizations for the levels of TMU that can be expected in actual measurement conditions. Many of the ranges for uncertainties are based on experience, as well as evaluations of variations in responses based on changes in modeling parameters.

4.5.1.1 Measurement Statistics and Activity Ranges. Measurement statistics are quantified by the analysis software. Therefore this value is automatically established. Measurement statistics are reported at the 1σ level in the assay report. Typical measurement statistics will range from ±1% to ±3% for high-activity items to approximately ±30% for activity at MDA levels. However, the total activity in the item is a factor in the overall uncertainty. This is because in addition to the actual measurement statistics, there are two other factors relating to the activity that contribute to the total measurement uncertainty.

For high-activity items, the analysis will have good quantitative values for the three strongest plutonium-239 gamma-energy lines (i.e., 129 keV, 375 keV, and 414 keV). This provides information that permits the analyst and final reviewer to compare the correction factors applied to each of the gamma lines. From the information, changes can be applied to the model when the activities indicate either an under-correction or over-correction of the gamma attenuation factors. Therefore, the uncertainties associated with matrix and absorbers are actually reduced.

For very low-activity measurements (typically in the LLW range), the plutonium-239 gamma-energy lines may be below detection levels. Therefore, the only detectable peak that can be used to calculate the content of the actinides in the item may be the 59.5 keV americium-241 line. Although this line has a high abundance, the low-energy gamma peak is significantly modified by changes in the absorbers or matrix elements, and also significantly affected by the assumptions of internal versus external contamination. The use of the low-energy americium-241 line to calculate the total TRU activity will be used only for light-weight items (typically less than 10 lb). When the americium-241 is used to calculate the total TRU activity, it will be assumed to be 30% of the total TRU activity to provide conservatism in the calculation. It may be noted that based on Table 4-5, americium-241 is 41.8% of the total TRU activity.

4.5.1.2 Source Location and Uniformity. Source location and uniformity may be accurately known in some cases for the 233-S Facility D&D items; however, in other cases the source location and uniformity may be unknown. Therefore, various types of items will be discussed based on the relative uncertainty in the source location(s) within the items, and assuming activity levels consistent with experience to date. In addition, as mentioned in the previous section, the uncertainty associated with source location is also related to gamma energy. Therefore, uncertainty approximations will be stated for 59.5 keV, 129 keV, and 414 keV gamma-energy lines, as shown in the following discussion.

4.5.1.2.1 Waste Items Containing Process Lines. For waste items containing process lines such as pipes, ducts, diffusers, etc., it can be assumed that except for some potential contamination on the outside during the decommissioning process, the bulk of the contamination should be on the inside. For most of these items where materials were passed through the system, deposition on the inside walls tends to be relatively uniform, except at joints or elbows.

Gamma Energy	Measurement Uncertainty
59.5 keV	±40%
129 keV	±20%
414 keV	±10%

4.5.1.2.2 Items Removed from Process Hood or Other Contamination Areas. Items removed from the process hood may be contaminated on the inside, outside, or both. Plates or other items that do not have interior spaces can obviously be assumed to have only external contamination; however the distribution of the contamination may or may not be uniform.

Items with Internal and External Surfaces:

Gamma Energy	Measurement Uncertainty
59.5 keV	±100%
129 keV	±40%
414 keV	±20%

Items with External Surfaces:

Gamma Energy	Measurement Uncertainty
59.5 keV	±25%
129 keV	±15%
414 keV	±10%

4.5.1.2.3 Liquids or Other Process Materials That Are Contained in Bottles, Jars, etc. This category of materials is assumed to have uniformly distributed activity in the container unless there is an indication of obvious separation of materials. However an unknown is whether the container was removed from an internal contamination area and may therefore have external contamination as well as (or instead of) internal contamination.

Gamma Energy	Measurement Uncertainty
59.5 keV	±25%
129 keV	±15%
414 keV	±10%

Currently the waste items have not been physically categorized (as stated in the earlier sections). Therefore, in all cases (except for items that do not have internal void spaces) the assumption is that the contamination is internal to the item being assayed rather than external. This will provide for a conservative estimate of the assay value.

4.5.1.3 Matrix, Absorber, Geometry, and Attenuation. Uncertainties associated with the geometry should be small because the physical dimensions of the package are easily established. Therefore, no direct uncertainty component will be applied for the overall geometry. The depth of the container (the dimension of the container that the detector is facing, ties into overall matrix uncertainties. Hence, for items that have an internal matrix attenuation component, there will be an uncertainty factor relating to the source distribution in the matrix.

There is a wide range of uncertainties associated with the matrix and absorber corrections. For well-defined geometries (e.g., pipes and ducts), the absorber can be easily defined. For bags containing a mixture of hard and soft waste, little information may exist as to the actual content of the package, therefore leading to significant uncertainties. As with the source distributions, the following discussion categorizes the levels of uncertainty associated with different types of items.

4.5.1.3.1 Pipes and Ducts. Pipes and ducts typically are standard components constructed from known materials, with known wall thickness. These are normally empty, so there is no internal matrix to consider as a part of the overall gamma absorption; therefore, a thin internal contamination level can be calculated.

For single items, the expected matrix uncertainty associated with items in this category is reported in the following table. In some cases, multiple pipes have been bundled together. In this case, the additional uncertainty associated with the multiple items can be assumed, as a first approximation, to be additive. Therefore, for example, a bundle of three pipes would have an absorber uncertainty of ±15% for the 414 keV gamma energy line.

Gamma Energy	Measurement Uncertainty
59.5 keV	±20%
129 keV	±10%
414 keV	±5%

4.5.1.3.2 Bags, Boxes, and Barrels of Waste. The uncertainty associated with matrix attenuation becomes a very complicated combination of factors associated with gamma energy, matrix density, and matrix depth. In general, average package densities (even those that contain high-density items, such as steel) are relatively low (<1 g/cc), which is due to void spaces or the mixture of "hard" and "soft" waste in one package. Therefore, the evaluation for matrix density will not be considered above 1 g/cc. The combination of energy, density, and depth is shown in Tables 4-14, 4-15, and 4-16 (one for each energy level) for matrix density versus matrix depth. These are crude estimates but are primarily to provide general ranges for potential measurement uncertainties.

Table 4-14. Measurement Uncertainty – Matrix Density vs. Depth for 414 keV Gamma-Energy Line.

Depth →	4 in.	8 in.	12 in.	16 in.	20 in.
Density ↓					
0.1 g/cc	±5%	±10%	±20%	±40%	±80%
0.3 g/cc	±10%	±20%	±40%	±80%	±160%
0.6 g/cc	±20%	±40%	±80%	±160%	±320%
1.0 g/cc	±40%	±80%	±160%	±320%	±640%

Table 4-15. Measurement Uncertainty – Matrix Density vs. Depth for 129 keV Gamma-Energy Line.

Depth →	4 in.	8 in.	12 in.	16 in.	20 in.
Density ↓					
0.1 g/cc	±10%	±20%	±40%	±80%	±160%
0.3 g/cc	±20%	±40%	±80%	±160%	±320%
0.6 g/cc	±40%	±80%	±160%	±320%	±640%
1.0 g/cc	±80%	±160%	±320%	±640%	±1,280%

Table 4-16. Measurement Uncertainty – Matrix Density vs. Depth for 59.5 keV Gamma-Energy Line.

Depth →	4 in.	8 in.	12 in.	16 in.	20 in.
Density ↓					
0.1 g/cc	±25%	±50%	±100%	±200%	±400%
0.3 g/cc	±50%	±100%	±200%	±400%	±800%
0.6 g/cc	±150%	±300%	±600%	±1,200%	±2,400%
1.0 g/cc	±500%	±1,000%	±2,000%	±4,000%	±8,000%

Tables 4-14, 4-15, and 4-16 imply totally unacceptable measurement uncertainties particularly for the lower gamma energies when attenuated in a thick high-density matrix. These uncertainties may be tempered by assumptions of source uniformity in the matrix. The following assumption would follow along the lines of normal sampling uncertainties.

For a larger container in which many individual items have been placed, the distribution of items with varying activities should be somewhat random (unless, of course, some packaging procedure would dictate placing higher dose rate items in the center of the container to minimize the overall container dose rate). Therefore, measurement of the portions of the container closest to the external surfaces represents a significant sampling of the overall content of the container.

As an example, consider a standard 55-gal drum. The 55-gal drum has a diameter of approximately 23 in. If the container is rotated during an assay, the detector will equally view all vertical surfaces of the drum, and if the detector is at a reasonable distance from the side of the drum (24 to 30 in.), then the variation in vertical response for the drum will be relatively small (25% at the extremes). Hence, assaying the outer 4-in. thickness of the drum represents the equivalent of sampling over 50% by volume.

Therefore, for most large items the uncertainty associated with the outer 4 in. of the container should be considered for the uncertainty analysis. This should then be multiplied by the percent of the volume.

4.5.1.4 Isotopic Ratios and Scaling Factors. Work at the 233-S Facility included processing materials with different plutonium isotopic ratios. In addition, different portions of the process would be expected to have somewhat varying ratios or scaling factors. Because many of the nuclide activities are calculated from the isotopic ratios or scaling factors, there is an uncertainty associated with the final results from using these factors. The magnitude of the uncertainty is very different depending on whether the values are being used for determining a gram value or for determining the radiological activity of the item.

Plutonium-238 and americium-241 are the primary contributors to the TRU activity for the 233-S Facility waste streams. Because the americium-241 activity can vary significantly relative to the plutonium-239 activity (based on evaluation of historical sampling/analysis data), the uncertainty in the overall isotopic ratio can cause a 30% to 50% uncertainty in the calculation of the total TRU activity value.

Although there are uncertainties associated with the isotopic ratios, the actual mass ratio for plutonium-239 will be between 86% and 94%. The total plutonium mass used for the criticality value will have an uncertainty associated with the isotopic ratios in the order of 5%. Therefore, the effect of uncertainty of the isotopic ratio on the total TRU activity with respect to criticality will be small relative to other uncertainties.

The actual scaling factor uncertainty would be summed in quadrature with the overall TMU to determine the uncertainty measurement for each nuclide.

4.5.2 Examples for Calculation of Total Measurement Uncertainty

For the total measurement uncertainty, assume all uncertainties are random and therefore multiply the uncertainties in quadrature. Hence, the following equation will be used to calculate the total measurement uncertainty:

$$TMU(1\sigma) = SQRT(\sigma meas\char`^2 + \sigma src\char`^2 + (\sigma att/\%vol)\char`^2) \tag{4-1}$$

where:

σmeas	=	uncertainty attributed to measurement statistics and activity range
σsrc	=	uncertainty attributed to source locations and uniformity
σatt	=	attributed to matrix, absorber, geometry, and attenuation
%vol	=	percentage of the total volume of the sample being considered for large volume samples

$$\text{TMU at 95\% confidence interval} = TMU(1\sigma) * 1.96 \tag{4-2}$$

The following examples show that an approximation of the overall measurement uncertainty can vary from quite reasonable to unreasonably large. However, in general for the gram quantity shots where the measurement uncertainty must be added for conservatism, the uncertainties are typically reasonable.

Example #1: 2-in. pipe with 1 g of plutonium-239:

There should be strong lines for all plutonium-239 peaks. Therefore, the 414 keV uncertainties will be applied. This might represent a typical gram shot.

Parameter	Values	Basis
σmeas	±2%	Industry experience
σsrc	±10%	Section 4.5.3.1
σatt	±5%	Section 4.5.4.1
%vol	100%	Entire pipe is "seen" by the detector
TMU (1σ)	±11.4%	Equation (4-1)
TMU @ 95% confidence	±22.3%	Equation (4-2)

Example #2: TRU characterization of same pipe:

In this case, it is assumed that only a detectable 129 keV plutonium-239 peak and a TRU concentration of approximately 200 to 300 nCi/g.

Parameter	Values	Basis
σmeas	±10%	Industry experience
σsrc	±20%	Section 4.5.3.1
σatt	±10%	Section 4.5.4.1
%vol	100%	Entire pipe is "seen" by the detector
TMU (1σ)	±24.5%	Equation (4-1)
TMU @ 95% confidence	±48.0%	Equation (4-2)

Example #3: same pipe with only americium-241 detectable:

Parameter	Values	Basis
σmeas	±10%	Industry experience
σsrc	±40%	Section 4.5.3.1
σatt	±20%	Section 4.5.4.1
%vol	100%	Entire pipe is "seen" by the detector
TMU (1σ)	±45.8%	Equation (4-1)
TMU @ 95% confidence	±89.8%	Equation (4-2)

Example #4: 8-in.-thick bag of assorted waste items from inside process hood (gram-level measurement, shot from both sides; average density 0.6 g/cc):

Parameter	Value	Basis
σmeas	±3%	Industry experience
σsrc	±20%	Section 4.5.3.2.1
σatt	±20%	Section 4.5.4.2, Table 4-14 (Note: The shot from each side can view the outer 4 in.; therefore, 4 in. on each side covers the full depth of the item.)
%vol	100%	Entire bag is "seen" by the detector
TMU (1σ)	±28.4%	Equation (4-1)
TMU @ 95% confidence	±55.7%	Equation (4-2)

Example #5: same bag as previous example but only 129 keV peak detectable:

Parameter	Value	Basis
σmeas	±10%	Industry experience
σsrc	±40%	Section 4.5.3.2.1
σatt	±40%	Section 4.5.4.2, Table 4-15 (Note: The shot from each side can view the outer 4 in.; therefore 4 in. on each side covers the full depth of the item.)
%vol	100%	Entire bag is "seen" by the detector
TMU (1σ)	±57%	Equation (4-1)
TMU @ 95% confidence	±112%	Equation (4-2)

Example #6: 55-gal drum with a density of 0.6 g/cc and gram level quantities of plutonium:

Parameter	Value	Basis
σmeas	±5%	Industry experience
σsrc	±20%	Section 4.5.3.2.1
σatt	±20%	Section 4.5.4.2, Table 4-14
%vol	50%	Section 4.5.4.2
TMU (1σ)	±45%	Equation (4-1)
TMU @ 95% confidence	±88%	Equation (4-2)

Example #7: same drum, but only 59.5 keV americium peak detectable:

Parameter	Value	Basis
σmeas	±10%	Industry experience
σsrc	±100%	Section 4.5.3.2.1
σatt	±150%	Section 4.5.4.2, Table 4-14
%vol	50%	Section 4.5.4.2
TMU (1σ)	±316%	Equation (4-1)
TMU @ 95% confidence	±620%	Equation (4-2)

4.6 STEP 7 – MEASUREMENT DESIGN

The following sections describe how the measurement program operates, measurement requirements, QC requirements, etc. Every waste item that is generated from the D&D of the 233-S Facility will undergo a radiological survey. The radiological survey could be performed by a direct-reading instrument or by an NDA system. The following subsections describe the NDA measurement process.

4.6.1 Measurement Process

The measurement process consists of five steps:

1. Measure and position sample.
2. Generate the model and the worksheet.
3. Perform gamma assay(s).
4. Generate report and enter information in the logbook.
5. Enter the results into the spreadsheet (validated and reviewed for quality assurance [QA]).

The measurements will be performed using a validated operating procedure.

4.6.2 Measurement Requirements

Prior to assaying items for the day, the operator must perform the control measurements as defined in Section 5.5.6.3. Measurements may only be performed when the system is "in control" (see Section 5.6.3).

Sample positioning must be appropriate for the item being assayed. The typical source to item distance is 24 in. The detector distance should be at least 50% of the longest item dimension. With this ratio, the detector will detect radiation from the full field of view of the item being measured (assuming perpendicularity at the center of the item). For longer items, multiple measurements along the length of the item should be performed. Small items (e.g., bottles or samples) can be placed closer to the detector but should never be closer than 25.4 cm (10 in.)

from the detector face to obtain a quantitative measurement. High-dose-rate items (i.e. > 10 mR/hr) can be placed farther from the detector to reduce the input count rate to the detector. Boxes will be "shot" in 1.2-m (4-ft) sections, with two "shots" (or measurements) on each side and one on each end. Thus, a 1.2-m by 1.2-m by 2.4-m (4-ft by 4-ft by 8-ft) box or 0.9-m by 1.2-m by 1.5-m (3-ft by 4-ft by 5-ft) box will have six shots. Larger size containers (e.g., boxes, drums, barrels, in-place equipment, and soils) will be evaluated on a case-by-case basis and a measurement plan for the item will then be developed.

The assay measurements must be performed for a long enough assay time to ensure that the measurement criteria can be met. For gram quantity items, this is typically a 5-minute assay. Measurements that are intended to generate the TRU radiological characterization or validate the waste as low-level are typically 30-minute assays. For items that are expected to have a uniform response (e.g., pipes, bottles, or plates), the 30-minute assay is divided up into a 15-minute front assay, combined with a 15-minute back assay. The operator should review the results to ensure that they meet the sensitivity of the measurement requirements. If the measurements are not satisfactory, then a longer assay time must be applied.

An item worksheet must be filled out for each item that is assayed. The worksheet must include the following information about the waste items: dimensions, weight, detector to item distance, description, material of construction, dose rate, measurement positions (if multiple measurements are performed) and any other relevant information that will aid in the analysis of the raw data.

PART II

QUALITY ASSURANCE PROJECT PLAN

Provides the quality assurance project plan, including the activities and guidelines to provide data of known and appropriate quality.

5.0 QUALITY ASSURANCE PROJECT PLAN

The information provided in this section (Part II) identifies the individuals or organizations participating in the project and discusses specific roles and responsibilities. The quality objectives for measurement data (laboratory analyses, NDA, and radiological surveys) and the special training requirements for the staff performing the work are also discussed.

5.1 PROJECT MANAGEMENT

The project shall be managed through the 233-S Facility Decommissioning Project that has an assigned project manager and project engineer. The project Field Support organization shall provide project assistance in performing radiological surveys and collecting samples for waste designation. The ERC Sample and Data Management organization shall arrange for analytical services. The ERC's Safety and Health organization shall provide radiological control and safety support, as required. The ERC Assessments, Regulatory, and Quality Programs organization shall be responsible for performing independent QA activities.

The NDA services will be performed by a specialty provider, who will report to the subcontract technical representative from the ERC's Environmental Technologies department. The NDA services provider will work closely with the project waste management representative to perform the NDA surveys for the waste items. The NDA services team will include a NDA project manager, a NDA data analyst, a NDA operator, and a QA representative.

5.2 *ROLES AND RESPONSIBILITIES*

This subsection identifies the responsibilities of various organizations supporting the waste characterization effort for designation and disposition. The 233-S D&D Project organizational chart can be found on the ERC Intranet (under "Functional Groups" and then "Human Resources"), which contains the most current ERC organizational information.

- D&D Characterization
  - Prepare the characterization plan
  - Arrange sampling and analysis activities
  - Oversee sampling
  - Interpret analytical data
  - Prepare the final characterization report.

- Project Engineering
  - Approve engineering calculations for development of isotopic ratios
  - Approve scaling factors to use for NDA and radiological surveys.

- Sample Management
  - Arrange for laboratory analysis of samples
  - Develop and issue SAF/field sampling request (FSR)
  - Receive data packages from the laboratory
  - Provide unique sample numbers for sample identification
  - Provide laboratory data package
  - Validate data to level identified in this plan
  - Provide subcontract technical representative for the NDA services contract.

- Field Sampling
  - Perform sampling and field screening
  - Provide certified clean sampling bottles/containers
  - Document sampling activities in a controlled logbook
  - Initiate chain-of-custody documentation for samples
  - Package and transport samples to the laboratory or shipping center.

- 233-S D&D Field Support
  - Prepare work packages to support the sample team
  - Conduct and document pre-job meetings when supporting the sample team
  - Provide field support to the sample team
  - Provide coordination with other site organizations (e.g., radiation control and safety) to support the sample/survey/NDA services team.

- Industrial Safety and Quality, Safety, and Health (QS&H)
  - Provide industrial safety support and monitoring for the sample team.
  - Provide the approved job hazard analysis (JHA).

  NOTE: The PPE to be worn during sampling shall be listed on the job-specific JHA or radiological work-permit (RWP), as required.

- Radiological Controls and QS&H
  - Provide radiation control coverage for the sample/survey/NDA services team
  - Provide dose rate data for sample collection, packaging, and shipping
  - Recommend ALARA actions where necessary
  - Provide RWPs
  - Conduct radiological surveys.

- BHI Waste Management/Transportation
  - Provide waste designation
  - Provide waste packaging instructions
  - Provide waste transportation.
- QA/QC and QS&H
  - Conduct routine assessments of Analytical Service providers
  - Conduct random surveillances to verify compliance with requirements of this quality assurance project plan (QAPjP).
- Data end-users
  - Project engineering
  - Field Support services
  - Radiological Controls
  - Design Engineering
  - Industrial Safety
  - Waste Management.
- NDA services provider
  - Provide NDA equipment
  - Operation of NDA equipment
  - Collect NDA measurements
  - NDA data reduction
  - NDA equipment maintenance
  - NDA measurement- specific QA/QC
  - NDA data verification and reporting

5.3 TRAINING REQUIREMENTS

Training or certification requirements needed by personnel are described in BHI-HR-02, *ERC Training Procedures*, and BHI-QA-03, *ERC Quality Assurance Program Plans,* Plan Nos. 5.1, 5.2, and 5.3. The Environmental Safety and Health Training Program also provides workers with the knowledge and skills necessary to safely execute assigned duties. Training programs of ERC subcontractors shall be approved by BHI. A graded approach is used to ensure that workers receive a level of training commensurate with their responsibility that complies with applicable DOE orders and government regulations. Specialized employee training includes pre-job briefings, on-the-job training, emergency preparedness, plan of the day, and facility/work site orientations, including all members of the building emergency response organization. The following training and qualifications will be applicable for the 233-S Facility work and environmental characterization activities, as appropriate:

- Training in accordance with 29 CFR 1910.120(e):
  - 40-Hour Hazardous Waste Worker/8-Hour Refresher
  - 24-hour experience component
  - 8-Hour Supervisor Training (for selected individuals)
  - Pre-job briefing.
- Other:
  - Respirator Training
  - First Aid (two qualified persons per shift/crew)
  - Certified Asbestos Worker/or Asbestos Awareness
  - Lead Worker Training
  - Samplers shall meet training requirements of BHI-EE-01, *Environmental Investigations Procedures*, Procedure 1.12, "Indoctrination, Training and Qualification."
- Training in accordance with the BHI-RC-01:
  - Radiation Worker II
  - Criticality Safety Training (site-specific).
- Medical surveillance requirements:
  - Hazardous waste worker physical
  - Respirator user medical
  - Mask fit
  - Lead worker baseline
  - Asbestos worker.
- Dosimetry and bioassay requirements:
  - Thermoluminescent dosimeter
  - Confirmatory plutonium urinalysis.
- Specialty training: Personnel performing NDA services will be trained in the operation and use of the NDA equipment. All personnel must meet the position requirements for their respective jobs. Operators must have completed the required reading and have a minimum of 40 hours of on-the-job training. Data analysts must have completed the required reading and have had a 24 hours of training on the operation of the equipment/modeling software.

5.4 DOCUMENTATION AND RECORDS

Sample collection and analysis activities shall be planned in accordance with BHI-EE-01, Procedure 1.5, "Field Logbooks"; Procedure 1.15, "Sampling Documents"; Section 2, "Sample Management"; and Section 3, "General Sampling." The SAF/FSR information generated through the sample event coordination process shall specify the sampling container, size, and preservatives; onsite measurements test methods; and laboratory analytical methods, turnaround times, and data deliverable types. Careful coordination with Radiological Protection and the laboratory is required to minimize sample volumes and potential radiological exposures associated with sample collection, packaging, and shipping.

The NDA measurements will be documented as required by the NDA subcontractor procedure or in a radiological survey record form or field logbook, as appropriate. Documentation includes a description of measurements performed, as well as notations of any adjustments to the system or abnormal occurrences. In addition, a formal worksheet is filled out for each waste item to be assayed. This includes the sample dimensions, sample weight, measurement distance, item description, positions of counts when multiple counts are performed, attenuation filters used, and any other information that will aid the analyst in evaluating the NDA measurement results.

Radiological surveys performed by the radiological control technicians will be documented in accordance with the requirements of BHI-RC-04, Instruction 4.2.

Field documents related to sample collection and sample packaging and shipment shall be maintained in accordance with BHI-EE-01, including the following procedures:

- Procedure 1.5, "Field Logbooks"
- Procedure 3.0, "Chain of Custody"
- Procedure 3.1, "Sample Packaging and Shipping."

5.5 MEASUREMENT DATA ACQUISITION

The following subsections present the quality objectives for measurement data and requirements for sampling methods, sample handling and custody, analytical methods, and field and laboratory QC. The requirements for instrument calibration and maintenance, supply inspections, and data management are also discussed.

5.5.1 Action Levels, Quality Objectives, and Criteria for Measurement Data

The action levels, quality objectives and criteria for measurement data are summarized in Tables 1-5a, 1-5b, 3-2, and 4-4, and in Section 4.5.1. Precision and accuracy requirements for analyses/assays are set by the analytical/assay methods used. Applicable performance requirements for radiological survey instrumentation are specified in BHI-RC-05, Instruction 2.1, "Operating Portable Instruments."

5.5.2 Sampling and Analysis Method Requirements

The sampling procedures to be implemented in the field will be consistent with those outlined in BHI-EE-01 and subcontract procedures meeting *Hanford Analytical Services Quality Assurance Requirements Document* (HASQARD) (DOE-RL 1996b).

5.5.3 Sample Handling and Custody Requirements

All sample handling, shipping, and custody requirements shall be performed in accordance with BHI-EE-01, Procedure 3.1, "Sample Packaging and Shipping"; Procedure 3.0, "Chain of Custody"; and Procedure 4.2, "Sample Storage and Shipping Facility."

5.5.4 Sample Preservation, Containers, and Holding Times

Sample preservation and container details will be addressed in the SAF/FSR in accordance with BHI-EE-01, Procedure 2.0, "Sample Event Coordination." The sample preservation, container, and holding times may be impacted by expected high TRU contaminant concentrations and resulting handling restrictions, potential requirements for laboratory or field extractions, etc. These items may adversely affect holding times for certain constituents and the ability to analyze for other constituents. If sample preservation, container type, or holding times cannot be met due to radiological contamination levels, this information shall be documented in the field logbook.

5.5.5 Laboratory Analytical Method Requirements

Samples will be sent to an ERC-approved laboratory that performs analyses in accordance with SW-846 guidelines. Methods requirements are identified in Table 3-2. The requirements for the project analytical needs are also defined in Table 3-2 by the call-outs for analytical technique, detection limits, and laboratory.

Table 3-2 describes sample volumes needed for each analysis requested. The sample volumes are separated into maximum volumes for full protocol analysis and minimum volumes for quick-turnaround data. Previous sampling activities at the 233-S Facility, process knowledge, and ALARA information indicate that under most circumstances maximum volume collection may not be achieved. Each sample location will be evaluated on a case-by-case basis to determine if full protocol will be used or if minimum volume collection will be used for quick-turnaround data.

5.5.6 Quality Control Requirements

The QC procedures must be followed in the field and laboratory to ensure that reliable data are obtained. When performing this field sampling effort, care shall be taken to prevent the cross-contamination of sampling equipment, sample bottles, and other equipment that could compromise sample integrity. Deviations shall be controlled in accordance with BHI-EE-01, Procedure 2.7, "Sample Disposition Record."

5.5.6.1 Quality Control Requirements for Sample Collection. The QC requirements for field sample collection activities are as follows:

- One duplicate sample, or a minimum of one field duplicate per every 20 samples of the same matrix, will be collected. Field duplicates are two samples produced from the same material and collected in the same location or from the same equipment. Field duplicates provide information concerning the homogeneity of the matrix, and an evaluation of the precision of the sampling and analysis process.

- Specific sampling instructions will be included in the work packages.

5.5.6.2 Quality Control Requirements for Radiological Surveys. The QC requirements for radiological surveys are provided in BHI-RC-04, Instruction 4.2 and BHI-RC-05, Instruction 2.1.

5.5.6.3 Quality Control Requirements for Nondestructive Assay Services. The QC requirements are a multi-tier program that cover all of the key aspects of the software, measurement process, analysis and reporting functions. The primary QC programs meet HASQARD requirements and are as follows:

- Software QA: The software used in this program will be a commercial software that has been designed and tested in accordance with the technical requirements. A verification and validation document is included with the software to document the performance capabilities. All spreadsheets used for final data manipulations are supported by design documents and validation test documents. Spreadsheets are controlled to ensure that the operator or analyst cannot modify calculations.

- Measurement control: A background measurement and a measurement control count will be performed by the operator at the beginning and end of each day. If any single-source measurement performed is outside allowable warning limits, the measurement will be immediately repeated a second time. If the second measurement is also outside the allowable warning limit, then personnel must cease measurements with that system and contact the NDA project manager (or designee). If any of the single-source measurements falls outside of an allowable action limit, then personnel must cease measurements with that system and contact the NDA project manager (or designee).

  Measurement control counts are automatically plotted on control charts by the analysis software. The control charts are included with batch reports. Measurements outside of an action level or multiple measurements outside of an investigation level require the initiation of a nonconformance report.

- Precision and accuracy of the measurements: The precision and accuracy of the item measurements are a function of the item geometry, radionuclide concentration, and item attenuators.

  The measurement precision is approximately 20% to 30% near the MDA levels and approximately 2% to 3% for the gram-level measurements.

  The measurement accuracy is primarily dependent on the knowledge and ability to accurately model the item being measured. Expected accuracy can be approximately based on the 1 σ TMU calculations.

- Independent technical review: All analyses are independently reviewed by another qualified analyst. This covers a review of the model, the analysis results, and a review for transcription errors. The final report is signed off by the project manager prior to release.

- Instrumentation calibration and maintenance: The detection system uses a high purity germanium detector that must have a valid efficiency calibration. Calibration standards must be traceable to a national or international standards laboratory.

5.5.7 Instrument/Equipment Testing, Inspection, and Maintenance Requirements

5.5.7.1 Radiological Surveys Using Portable Instruments. With the exception of radiological surveys performed in accordance with BHI-RC-04, all field screening and analytical instruments shall be tested, inspected, and maintained in accordance with BHI-QA-03, Plan No. 5.2, "Onsite Measurements Quality Assurance Program Plan," and Plan No. 5.3, "Environmental Radiological Measurements Quality Assurance." The results from all testing, inspection, and maintenance activities shall be recorded in a bound logbook in accordance with procedures outlined in BHI-EE-01, Procedure 1.5, "Field Logbooks."

With the exception of radiological surveys performed in accordance with BHI-RC-04, all field screening and analytical instruments shall be calibrated in accordance with BHI-QA-03, Plan No. 5.2. The results from all instrument calibration activities shall be recorded in a bound logbook in accordance with BHI-EE-01, Procedure 1.5. Tags will be attached to all field screening and onsite analytical instruments, noting the date when the instrument was last calibrated along with the calibration expiration date. Any discovered calibration deficiencies shall be documented.

General requirements for setup and operation of hand-held instruments are provided in BHI-RC-05, Instruction 2.1, "Operating Portable Instruments."

5.5.7.2 Nondestructive Assay Services. For NDA services, instrument calibration and maintenance requirements are as follows:

- The detection system to be used for the measurements uses a high purity germanium detector that must have a valid efficiency calibration. Sources that are National Institute of Standards and Technology (NIST)-traceable (or international standards traceable to NIST) must be used in the efficiency calibration process.

- The system operation must be monitored through a measurement control program (as previously described). This procedure monitors the operating characteristics of the system to ensure that the system is operating within reasonable statistical limits and is, therefore, "in control." These measurements are performed using a NIST-traceable (or international standards traceable to NIST) source (e.g. europium-152) that can also be used as a calibration validation source. If the system can not be kept in control then the system must be repaired and recalibrated. As long as the system can be demonstrated to be in control or revalidated using traceable sources, the original efficiency calibration is considered to be valid. If the system is repaired or components replaced in the system, a revalidation of the system performance must be performed at a minimum.

- Validation measurements should be performed annually with a traceable source that includes plutonium.

5.5.8 Inspection/Acceptance Requirements for Supplies and Consumables

Procurement activities will be limited to providing BHI Procurement with procurement requisitions. All subject activities will meet the requirements of BHI Procurement procedures found in BHI-PR-01, *ERC Procurement Procedures*.

The project will review received items and reagents for conformation to the specifications set in the requisition. If the item or reagent does not meet specifications, the item or reagent will be dispositioned through the nonconformance system.

The acceptability of new standards will be determined by comparing the new standard with previous acceptable standard(s). Reagent acceptability will be determined by running blanks on the new reagents. New reagents and standards will be separated from other standards and reagents until they have been checked and accepted.

5.5.9 Data Management

Data resulting from the implementation of the SAP will be managed and stored by the ERC's Sample Management organization in accordance with BHI-EE-01, Section 2.0, "Sample Management."

All validated reports and supporting analytical data packages shall be subject to final technical review by qualified reviewers before submittal to regulatory agencies or inclusion in reports or technical memoranda, at the direction of the BHI project task lead. Electronic data access, when appropriate, shall be through computerized databases (i.e., the Hanford Environmental Information System). Where electronic data are not available, hard copies will be provided in accordance with Section 9.6 of the *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) (Ecology et al. 1998).

All validated reports and supporting analytical data packages will be retained and dispositioned in accordance with BHI-MA-02, *ERC Project Procedures*, Procedure 1.7, "ERC Records Management."

5.6 ASSESSMENT/OVERSIGHT

5.6.1 Assessments and Response Actions

The QS&H quality service engineer may conduct random surveillance and assessments in accordance with BHI-SH-06, Procedure 3.1, "Surveillance," to verify compliance with the requirements outlined in the SAP, project work packages, the ERC Quality Program, BHI procedures, and regulatory requirements.

Deficiencies identified by self-assessments shall be reported in accordance with BHI-MA-02, *ERC Project Procedures*, Procedure 2.7, "Self-Assessment." When appropriate, corrective actions will be taken by the Project Engineer in accordance with the HASQARD, Vol. 1, Section 4.0 (DOE-RL 1996b), to minimize recurrence.

5.6.2 Reports to Management

Management shall be made aware of all deficiencies identified by the surveillances and self-assessments, and the deficiencies shall be reported in accordance with BHI-MA-02, Procedure 2.7 and BHI-SH-06, Procedure 3.1.

5.7 DATA VALIDATION AND USABILITY

5.7.1 Data Review, Validation, and Verification Requirements

Data verification and validation is performed on analytical data sets, primarily to confirm that sampling and chain-of-custody documentation is complete, sample numbers can be tied to the specific sampling location, samples were analyzed within the required holding times, and analyses met the data quality requirements specified in the field sampling plan (which is included as Part III of this SAP).

5.7.2 Validation and Verification Methods

All data verification and validation shall be performed in accordance with BHI-EE-01, Procedure 2.5, "Data Package Validation Process"; *Data Validation Procedure for Chemical Analysis* (BHI 2000c); and *Data Validation Procedure for Radiochemical Analysis* (BHI 2000d). A validation performed in a comparable manner to Level C, as described in the identified procedures, will be performed on onsite laboratory analyses. This allows the review of all QC data, transcription error verification, and holding time review. This level is the middle validation level and does not require review of raw data and recalculation of data. Should problems arise from the Level C review, the project reserves the option to review or recalculate.

5.8 RECONCILIATION WITH USER REQUIREMENTS

A data quality assessment will be performed on the resulting analytical data in accordance with *Guidance for Data Quality Assessment* (EPA 1996). The data quality assessment will determine if the data are the right type, quality, and quantity to support the intended use. The data evaluation for this project entails the following:

- Reviewing analytical data, including data packages and QA reports
- Drawing conclusions from the data
- Interpreting and communicating the test results.

PART III

FIELD SAMPLING PLAN

Provides field procedures to ensure representative data of known quality.

6.0 FIELD SAMPLING PLAN

6.1 SAMPLING OBJECTIVES

The objective of the field sampling plan is to clearly identify the sampling and analysis activities, NDA measurements and radiological surveys needed to resolve the DRs identified in Step 5 of the DQO summary report (Section 3.0). The FSP takes the sampling design proposed in Step 7 of the DQO summary report (Section 3.0) and presents the parameters to identify sampling locations, total number of samples to be collected, sampling procedures to be implemented, analyses to be performed, and sample bottle requirements.

6.2 SAMPLING LOCATIONS

The field sampling will be conducted using a phased approach. The first step will be visual inspection to identify accessibility, sample matrix, and sample volume. Exact sample locations will be confirmed with the D&D Characterization team members, sample personnel, and radiological control technician supervisors. The second step will be radiological surveys or NDA of specified locations. These locations will identify hot spot (or worst-case locations) for sampling. The third (and final) step will be sample collection for laboratory analysis. Table 3-3 describes the sample location, sample strategy, and sampling method. If ALARA reasons or field conditions prevent the collection of samples, as identified in Table 3-3, any deviations shall be documented in the field logbook.

Throughout the duration of the project, facility conditions will change and/or additional information will become available, which may alter the initial characterization plans. Uncertainties, such as the use of sampling equipment and accessibility, are possible. Therefore, the key to success of this characterization effort lies with the ability to adjust efforts in the field to appropriately react to the uncertainties or changing conditions.

6.2.1 DOE and EPA Approval of Specific Sample Events, Sampling Location, and Disposal

The D&D activities in the 233-S Facility are planned in a sequence that proceeds from areas of relatively low risk to areas of higher risk. Individual work packages will be used for sequential scopes of work. Sampling and characterization hold points in these work packages will allow for appropriate decision making.

When proposed sample locations have been identified, an electronic mail message will be sent to the DOE Assistant Manager for Environmental Management (DOE-AME) 233-S Program Manager identifying sample points, special sampling equipment and sample analyte priorities if there is not enough sample volume to run all analyses. Detection limits and precision and accuracy requirements would also be identified if they differ from those identified in the HASQARD (DOE-RL 1996b). Upon DOE's concurrence, the message will be electronically forwarded to the EPA for approval. Upon receipt of EPA's approval, the document will be entered into the BHI Document Information System database, which will assign a document number to the approved message for future tracking.

The primary disposal option, as identified in the action memorandum for the 233-S Plutonium Concentration Facility, for each waste stream is the ERDF waste acceptance criteria (BHI 1998). EPA approval is required for disposal of waste in locations other than the ERDF.

6.3 LABORATORY ANALYSIS

The COCs, analytical method, technique, required detection limit, and laboratory detection limits needed to support data for waste designation are summarized in Table 3-2. These analyses will support the identified waste streams as well as anomalies found during decommissioning activities. The sample volumes are separated into maximum volumes for full protocol analysis and minimum volumes for quick-turnaround data. Previous sampling activities at the 233-S Facility, process knowledge, and ALARA information indicate that under most circumstances, maximum volume collection may not be achieved. Each sample location will be evaluated on a case-by-case basis to determine if full protocol will be used or if minimum volume collection will be used for quick-turnaround data.

6.4 RADIOLOGICAL SURVEYS

Radiological surveys, using hand-held instruments, will be performed in accordance with BHI-RC-04, Instruction 4.2, and BHI-RC-05, Instruction 2.1 and will be used to identify total and removable contamination levels, providing dose-rate information and provide data/information for making waste management decisions. Additionally, radiological surveys may be performed prior to sample collection to identify areas of "worst-case" radiological contamination of the waste stream matrices. The amount of removable material per 100 cm^2 of surface area should be determined by wiping an area of that size with a dry filter or soft absorbent paper, applying moderate pressure, and then measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area <100 cm^2 is necessary, the activity per unit area should be based on the actual area, and the entire surface area should be wiped.

The amount of total contamination (removable plus fixed) should be determined by using appropriate instrument of known efficiency, and placing the probe of the instrument adjacent to the surface being surveyed. Use care when checking uneven surfaces.

Dose-rate surveys should be determined by using an appropriate instrument of known efficiency and taking readings on contact and at 30 cm from the item being surveyed. Information annotated on survey forms should indicate highest reading.

Survey data in the form of direct reading survey measurements, smear surveys, and dose-rate surveys will be used to verify process knowledge that the radioactive waste is LLW and is within the approved waste profile. The information obtained from the surveys will be recorded on a

BHI Radiological Survey Record. For conservatism, the highest levels (contamination and dose-rate information) indicated on the survey record will be used for waste verification purposes. This information will then be converted from the reported units (e.g., dose rate, disintegrations per minute) to an activity per unit mass. The basis for the conversion will be documented in a calculation performed in accordance with BHI-DE-01, *Design Engineering Procedures Manual*, Engineering Department Project Instruction (EDPI) 4.37-01. Examples of this conversion can be found in the following calculations:

- 0200W-CA-N0032, *233-S Determination of Step-Off Pad Waste Alpha Activity Concentration* (BHI 2001b)

- 0200W-CA-N0033, *233-S Determination of Soft Waste Alpha Activity Concentration* (BHI 2001a).

All radiological instruments used will be calibrated within the frequency specified in the instrument operating procedures. Daily instrument response checks for portable instruments will be performed in accordance with BHI-RC-05, Instruction 2.1.

6.5 NONDESTRUCTIVE ASSAY

As stated in Section 4.6, NDA will be used to provide data for designating the radiological component of waste items.

6.6 SAMPLING PROCEDURES

The sampling procedures to be implemented in the field shall be consistent with the procedures outlined in BHI-EE-01.

6.7 SAMPLE MANAGEMENT

Sample management activities shall be performed in accordance with the following BHI-EE-01 procedures:

- Procedure 3.1, "Sample Packaging and Shipping "
- Procedure 4.2, "Sample Storage and Shipping Facility"
- Procedure 3.0, "Chain of Custody."

6.8 WASTE MANAGEMENT

All waste (including unexpected waste) generated by sampling activities will be managed in accordance with BHI-EE-10, *Waste Management Plan* and the project waste management plan (provided in Appendix E of the 233-S removal action report [BHI 2000e]). Unused samples and associated laboratory waste for the analysis will be dispositioned in accordance with the laboratory contract and agreements for return to the Hanford Site. Pursuant to 40 CFR 300.440, Remedial Project Manager (RPM) (i.e., EPA Project Manger) approval is required before returning unused samples or waste from offsite laboratories.

In addition, RPM approval is required before shipping sample waste from Hanford onsite laboratories (e.g., 222-S Analytical Laboratories, Waste Sampling and Characterization Facility, Radiological Counting Facility, or Radiochemical Processing Laboratory) back to the waste site of origination. Approval of this SAP constitutes the RPM's approval of this action.

6.9 HEALTH AND SAFETY

All field operations will be performed in accordance with BHI health and safety requirements outlined in BHI-SH-01, *ERC Safety and Health Program*. In addition, a work control package will be prepared in accordance with BHI-MA-02 that will further control site operations. This work package will include an activity hazard analysis and site-specific health and safety plan, and will also reference applicable RWPs.

The sampling procedures and associated activities will take into consideration exposure reduction and contamination control techniques that will minimize the radiation exposure to the sampling team as required by BHI-RC-01 and BHI-QA-01, *ERC Quality Program*.

7.0 REFERENCES

10 CFR 61.55, "Licensing Requirements for Land Disposal of Radioactive Waste, Waste Classification," *Code of Federal Regulations*, as amended.

29 CFR 1910, "Occupational Safety and Health Standards," *Code of Federal Regulations*, as amended.

40 CFR 61, "National Emission Standards for Hazardous Air Pollutants," *Code of Federal Regulations*, as amended.

40 CFR 261, "Identification and Listing of Hazardous Waste," *Code of Federal Regulations*, as amended.

40 CFR 268, "Land Disposal Restrictions," *Code of Federal Regulations*, as amended.

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